Exam Bank No.: 2212 **Last used on an NRC exam:** Never

RO Sequence Number: 1

In accordance with 0POP01-ZA-0024, Enhanced Off Normal Operating Procedure Users Guide, which of the following is a responsibility of the Reactor Operators during the performance of an Off Normal Procedure?

A. Ensuring briefings are performed at appropriate transitions or pauses.

B. Predetermining manual actuation setpoints to be used when slowly degrading parameters are unrecoverable.

C. Performance of all immediate actions from memory.

D. Monitoring Conditional Information Pages for possible required actions.

Answer: C Performance of all immediate actions from memory.

Exam Bank No.: 2212

K/A Catalog Number: G2.4.11 **Tier:** 3 **Group/Category:** 4

RO Importance: 4.0 **10CFR Reference:** 55.41(b)(10)

Knowledge of abnormal condition procedures.

STP Lesson: LOT 505.02 **Objective Number:** 92114

GIVEN a job position, STATE the responsibilities associated with that position as stated in 0POP01-ZA-0024, Enhanced Off Normal Operating Procedure Users Guide.

Reference: POP01-ZA-0024, section 2.0

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because it is listed in the procedure as a responsibility for the SM and US.

B: INCORRECT: Credible because it is listed in the procedure as a responsibility for the SM and US.

C: CORRECT: This is a listed responsibility for the RO.

D: INCORRECT: Credible because it is a listed responsibility for the US.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have a knowledge of the Reactor Operators responsibilities durng Off Normal procedure performance.

1.0 Purpose

The purpose of this procedure is to provide guidance for consistent implementation of STPEGS Enhanced Off-Normal Procedures (ONPs). Enhanced Off-Normal Procedures are identified by a statement in the Cover Page heading identifying the procedure as an Enhanced Off-Normal Procedure.

Enhanced ONP usage is intended to be standardized with 0POP01-ZA-0018, Emergency Operating Procedure Users Guide, dual column format rules of usage. The two users guides are to be combined in the future, therefore the user is referred to 0POP01-ZA-0018 when appropriate rather than duplicate information.

2.0 Responsibilities

2.1 Shift Manager (SM) is responsible for:

- The overall implementation of the Enhanced ONPs.
- Directing all plant personnel actions per the Enhanced ONPs.
- Maintaining a broad perspective of events during the implementation of the Enhanced ONPs.
- Predetermining manual actuation setpoints to be used when slowly degrading parameters are unrecoverable AND determining when to abandon attempts to recover degrading parameters and manually initiate safety systems.
- Overview and general direction of recovery actions in accordance with station procedures.
- Implementing the Emergency Plan as required by plant conditions.
- Ensuring briefings are performed at appropriate transitions or pauses to review plant status, Emergency Plan classification, etc.

2.2 Unit Supervisor (US) is responsible for:

- Entering the Enhanced ONPs as warranted by plant conditions.
- Directing all plant personnel actions per the Enhanced ONPs.
- Monitoring Conditional Information Pages for possible required actions.
- Predetermining manual actuation setpoints to be used when slowly degrading parameters are unrecoverable AND determining when to abandon attempts to recover degrading parameters and manually initiate safety systems.
- Ensuring briefings are performed at appropriate transitions or pauses to review plant status, Emergency Plan classification, etc.

2.3 Reactor Operators (RO) are responsible for:

- The performance of all Enhanced ONP immediate actions from memory.
- The performance of all other Control Room actions as directed by the US.
- Communicating the completion or outcome of actions to the US.
- Communicating parameter trends to the US.
- Directing Reactor Plant Operators (RPO) actions per the Enhanced ONPs.

2.4 Reactor Plant Operators (RPO) are responsible for:

- The performance of all local actions as directed by the US or ROs.
- Communicating the completion or outcome of actions to the US or ROs.
- Communicating any abnormal local plant conditions to the US.

3.0 Procedure Hierarchy

3.1 Procedure use follows the normal convention for plant operating procedures, with the highest priority procedure being the primary procedure in use. The procedure priority, with the exception when stated otherwise, is:

3.1.1	Emergency Operating Procedures	(EOP)	(0POP05's)
3.1.2	Off Normal Operating Procedures	(ONP)	(0POP04's)
3.1.3	Annunciator Response Procedures	(ARP)	(0POP09's)
3.1.4	Normal Operating Procedures	(OP)	(0POP02's, 0POP03's)

Exam Bank No.: 2207 **Last used on an NRC exam:** Never

RO Sequence Number: 2

An audible alarm is received on the RM-11 system for RT-8035, FHB Exhaust.

Which of the following describes the correct response by the Reactor Operator to this condition?

A. Silence the alarm at the RM-11 console and then contact Health Physics to determine subsequent actions.

B Silence the alarm at the RM-11 console and then use POP04-RA-0001, Radiation Monitoring System Alarm Response, to determine subsequent actions.

C. Silence the alarm at CP-023 (RM-23) and then contact Health Physics to determine subsequent actions.

D. Silence the alarm at CP-023 (RM-23) and then use POP04-RA-0001, Radiation Monitoring System Alarm Response, to determine subsequent actions.

Answer: B Silence the alarm at the RM-11 console and then use POP04-RA-0001, Radiation Monitoring System Alarm Response, to determine subsequent actions.

Exam Bank No.: 2207

K/A Catalog Number: G2.3.13 **Tier:** 3 **Group/Category:** 3

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(12)

Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc

STP Lesson: LOT 505.01 **Objective Number:** 92105

STATE the purpose of, and DESCRIBE the scope of the referenced procedure.

Reference: 0POP04-RA-0001, page 3 (Purpose)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because Health Physics are the experts for radiation related matters, however the operators are expected to use the off-normal procedure for system related responses.

B: CORRECT: The audible alarm is silenced with the SYSTEM ACK key on the RM-11 keyboard. A radiation monitor alarm on RM-11 is an entry condition for POP04-RA-0001 which then provides diagnostics/actions for the operator.

C: INCORRECT: Credible because CP-23 does have various controls and indications for this monitor, but no way to silence the alarm. Health Physics are the experts for radiation related matters, however the operators are expected to use the off-normal procedure for system related responses.

D: INCORRECT: Credible because CP-23 does have various controls and indications for this monitor, but no way to silence the alarm.

Question Level: F **Question Difficulty** 3

Justification:

The applicant requires a knowledge of the purpose of the off-normal procedure for radiation monitoring and an understanding of how the RM-11 system functions.

PURPOSE

This procedure provides instructions and guidance to Control Room personnel to determine the cause of an ALERT or HIGH Area Process and Effluent Radiation Monitoring System alarm, evaluate the operability of the monitor or perform the initial required actions when the alarm is from the following panels and locations:

- RM-11, or RM-23A (when RM-11 is out of service) radiation monitoring panels.
- N16 Primary to Secondary Leak Monitoring System Alarms on the Plant Computer.

This procedure is applicable to all radiation monitors to provide the initial response to an alarm. However procedure 0PGP03-ZA-0078, Administration of the Radiation Monitoring System, provides the assignment of primary user and responsibility for the individual monitors.

The RM-11 system essentially provides the equivalent of an annunciator alarm display for the radiation monitors. The user may apply the rules for responding to an annunciator alarm when responding to alarms received on the RM-11 system.

SYMPTOMS OR ENTRY CONDITIONS

1. Alert, High or Status alarm on any of the following Radiation Monitors:

    - RT-8010A and RT-8010B, Unit Vent Stack
    - RT-8011, RCB Atmosphere
    - RT-8012 and RT-8013, RCB Purge Exhaust
    - RT-8014, RT-8015, RT-8016, RT-8017, RT-8018, RT-8029 and RT-8030, MAB Ventilation
    - RT-8022, RT-8023, RT-8024 and RT-8025, Steam Generator Blowdown
    - RT-8027, Condenser Air Removal System
    - RT-8031, GWPS Inlet
    - RT-8032, GWPS Outlet
    - RT-8033 and RT-8034, EAB Air Intake
    - RT-8035 and RT-8036, FHB Exhaust

Step 1. continued on next page

Exam Bank No.: 2208 **Last used on an NRC exam:** Never

RO Sequence Number: 3

Prior to a containment entry at power, Health Physics has requested the Containment Carbon Filter Units be placed in service.

Which of the following describes the purpose for this action?

A. The High Efficiency Particulate Air (HEPA) Filter contained within the unit will remove airborne radioiodine which is mainly an internal dose hazard.

B. The Charcoal Filter contained within the unit will remove airborne radioiodine which is mainly an internal dose hazard.

C. The High Efficiency Particulate Air (HEPA) Filter contained within the unit will remove airborne radioiodine which is mainly an external dose hazard.

D. The Charcoal Filter contained within the unit will remove airborne radioiodine which is mainly an external dose hazard.

Answer: B The charcoal filter contained within the unit will remove airborne radioiodine which is mainly an internal dose hazard.

Exam Bank No.: 2208

K/A Catalog Number: G2.3.12 **Tier:** 3 **Group/Category:** 3

RO Importance: 3.2 **10CFR Reference:** 55.41(b)(12)

Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high radiation areas, aligning filters, etc.

STP Lesson: LOT 202.33 **Objective Number:** 92035

DESCRIBE the flowpath and STATE the functions for each of the following RCB-HVAC subsystems:

B. Containment Carbon Units

Reference: LOT202.33 handout section 3.2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because the unit has a HEPA filter which is designed to remove radioactive particles, so the applicant must know the design basis for each.

B: CORRECT: The charcaol filter is there to remove iodine which is an internal hazard since it concentrates in the thyroid.

C: INCORRECT: Credible because the unit has a HEPA filter which is designed to remove radioactive particles, so the applicant must know the design basis for each. The radiation hazard is also credible since some radioactive isotopes are more an external hazard than internal.

D: INCORRECT: Credible because some radioactive isotopes are more an external hazard than internal.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have an understanding of the purpose of a charcoal filter and the hazards of radioactive iodine.

2.Containment Carbon Units Subsystem

A.The Containment Carbon Unit Subsystem consists of two 50 percent capacity units. Each unit consists of the following components:

a.Prefilters

b.HEPA filters (two banks - one upstream and one downstream of the carbon filters)

c.Carbon filters (rechargeable type)

d.Fans (two 100 percent capacity)

B.Prefilters

The prefilters are provided upstream of the HEPA filters to protect them from coarse particles, and are designed for 85 percent efficiency.

C.HEPA Filters

HEPA filters are provided to remove fine particulate matter from the airstream. HEPA filters are provided downstream of the carbon filters to collect any carbon fines which may be carried into the airstream from carbon filters.

D.Carbon Filter

The carbon filters are provided to remove the airborne radioiodine from the airstream. These can be exhausted by solvent vapors and other halogenated materials.

E.Circulating Fans (2)

The circulating fans are centrifugal type with direct-drive, single-speed motors. Fans have totally enclosed, air-cooled motors, and are statically and dynamically balanced.

F.Design Criteria

The Containment Carbon Units System shall reduce the airborne radioactivity levels in the containment atmosphere prior to allowing personnel access during normal plant operation or in advance of a scheduled reactor shutdown. It is not required for safe shutdown.

Rated flow through each carbon unit is 10,000 cfm.

Exam Bank No.: 19 **Last used on an NRC exam:** 2005

RO Sequence Number: 4

Unit 1 was at 100% power when a Large Break LOCA occurred.

Essential Cooling Water (ECW) Pump 1B was in Auto and started but the ECW Pump 1B discharge valve stopped at 80% open due to mechanical binding.

Which of the following describes the final condition of ECW Pump 1B and the reason why?

ECW Pump 1B…

A. remains running to continue cooling ESF support systems.

B. trips to protect pump casing from over pressure.

C. remains running to continue flow to the Screen Wash Booster Pump.

D. trips to protect pump shaft bearings from overheating.

Answer: A remains running to continue cooling ESF support systems.

Exam Bank No.: 19

K/A Catalog Number: APE 062 AK3.02 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(7)

Knowledge of the reason for the following responses as they apply to the Loss of Nuclear Service Water: The automatic actions (alignments) within the nuclear service water resulting from the actuation of the ESFAS.

STP Lesson: LOT 201.13 **Objective Number:** 91193

LIST all automatic functions, switch locations, switch positions, annunciators, local/remote functions, interlocks and permissive for the following: A) Traveling Screens, B) Screen Wash Booster Pump, C) Screen Wash Valve, D) Strainers, E) Pumps and Motors, F) Discharge Valve, G) Sump, H) Blowdown Valve, I) Sump Pump and Motor

Reference: 9E-EW01-01 Rev 19, 9E-EW04-02 Rev 12

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT - An SI actuation will block the trip of an ECW pump from discharge valve position even though a partial loss of the system may occur.

B: INCORRECT - Credible because a partially closed discharge valve would result in a higher discharge pressure which would raise the stress on the pump casing.

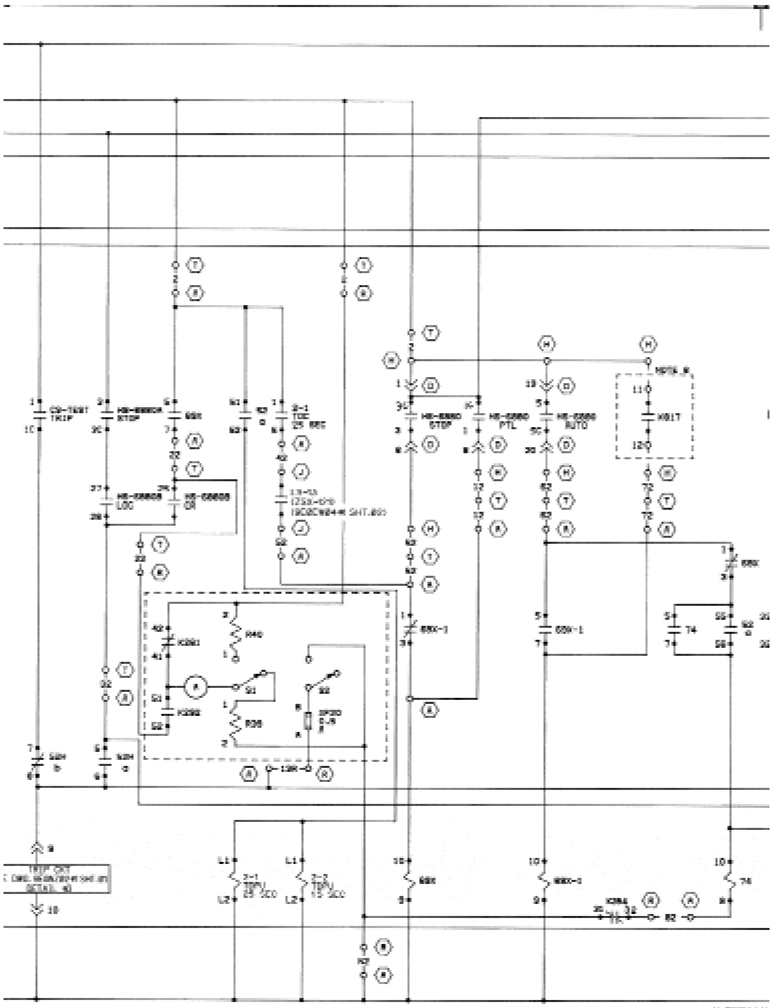
C: INCORRECT - Credible because the ECW pump supplies water to the booster pump, however that is not the reason why the ECW pump trip is blocked during an SI.

D: INCORRECT - Credible because the pump bearings are cooled and lubricated by the discharge flow of the pump and a partially closed valve will reduce the discharge flow.

Question Level: F **Question Difficulty** 3

Justification:

Applicant must have fundamental knowledge of the reasons for ECW Pump logics.

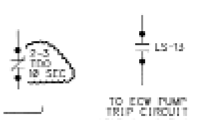


IAGRAM 2

R04C2
G WATER PUMP
GE
137

CABLE BLOCK DIAGRAM 3

SCHEME CEW04C3
ESSENTIAL COOLING WATER PUMP
DISCHARGE
CEW0MOV0151



TO ECW PUMP
TRIP CIRCUIT
9E0EW01#1 SHT. 01

REFERENCE DRAWING:

4020--01050#M	MOV WIRING DIAGRAM
4020--07414#M	MOV OUTLINE
00009E0MAJ#1 SHT.01	S/L DIAG 480V MCC E1C3
00009E0MAF#1 SHT.01	S/L DIAG 480V MCC E1B3
00009E0MAAC#1 SHT.01	S/L DIAG 480V MCC E1A3
5Z349Z47504 SHT. 10-3	SWITCH DEV & WIRING
5N289F05038#M	P&ID DIAGRAM
5N289Z4208#M	LOGIC DIAGRAM
0000G00007-0002206	BILL OF MATERIAL (E1A3-F3)
0000G00007-0002306	MCC INT WIRING (E1A3-F3)
0000080003-0002806	MCC INT WIRING (E1B3-D4)
0000080003-0003806	BILL OF MATERIAL (E1B3-D4)
0000080007-0003606	BILL OF MATERIAL (E1C3-D1)
0000080007-0003706	MCC INT WIRING (E1C3-D1)

CAB CABLE BLOCK DIAGRAM, SEE DWG 00009E0MR07#1 SHT. 01, (00009E0MR07#1 SHT. 02) [00009E0MR07#1 SHT. 03].

3. CABLE NUMBERS FOR POWER CABLE ARE ASSIGNED FROM REFERENCED ONE LINE DRAWINGS.

4. DELETED

5. ELEMENTARY DIAGRAM SHOWN IS FOR MOV-0121, MOV-0137 & MOV-0151 ARE SIMILAR EXCEPT AS NOTED ON BLOCK DIAGRAM & IN () BRACKETS.

6. DELETED

7. DELETED.

8. FOR MOTOR AND LIMIT SWITCH COMPARTMENT SPACE HEATER POWER FEED REFER TO DWG 00009E0VF02#1 SHT. 01.

NUCLEAR SAFETY RELATED | Q

SOUTH TEXAS PROJECT
NUCLEAR OPERATING COMPANY

ELEMENTARY DIAGRAM
ESSENTIAL COOLING WATER PUMPS-TRAINS A, B & C
DISCHARGE MOV'S 0121, 0137 & 0151

ORIG	CKR	RE	DV	SE	PRIORITY	CADD FILE NO.	PROJ. NO.	SIT. NO.	SCALE	DWG. NO.	SHT.	REV.
					1	E0161.121	D070026	33099119	NONE	00009E0EW04#1	02	12

4 | 3 | 2 | 1

Exam Bank No.: 52 **Last used on an NRC exam:** 2005

RO Sequence Number: 5

Given the following Unit 2 conditions:

- A Small Break LOCA has occurred
- SI has been reset
- Operators have just completed step 1 of 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant

The Shift Technical Advisor reports the following:

- RCS pressure:........................1830 psig
- RCS Subcooling:....................60^oF
- Pressurizer level: 20%
- SG A NR Level:.....................8%
- SG B NR Level: 10%
- SG C NR Level: 17%
- SG D NR Level:.....................19%
- Total AFW flow:...................400 gpm
- Adverse containment conditions do NOT exist

Which of the following actions should the operators perform?

A. Manually actuate SI and transition to 0POP05-EO-EO00, Reactor Trip or Safety Injection

B. Transition to 0POP05-EO-FRH1, Response to Loss of Secondary Heat Sink

C. Transition to 0POP05-EO-FRI2, Response to Low Pressurizer Level

D. Transition to 0POP05-EO-ES11, SI Termination

Answer: D Transition to 0POP05-EO-ES11, SI Termination

Exam Bank No.: 52

K/A Catalog Number: EPE W/E02 EA2.2 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(10)

Ability to determine and interpret the following as they apply to the (SI Termination): Adherance to appropriate procedures and operation within the the limitations in the facility's license and amendments.

STP Lesson: LOT 504.09 **Objective Number:** 81187

DISCUSS the indications available to determine plant status during a loss of primary or secondary coolant accident.

Reference: 0POP05-EO-EO10, Rev 21 Conditional Information Page

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible since SI reinitiation criteria would be met if adverse containment conditions existed.

B: INCORRECT - Credible because this transition is required if adverse containment conditions existed or if all SG levels were less than 14%.

C: INCORRECT - Credible because transition may be done with a lower pressurizer level (17%).

D: CORRECT - The given conditions would allow transition to ES11 which would be the expected action for the Crew.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must analyze the given conditions and apply their knowledge of SI termination and reinitiation requirements and the loss of heat sink and integrity transitions in order to eliminate the incorrect responses and choose the correct response.

CONDITIONAL INFORMATION PAGE

SI TERMINATION CRITERIA

IF All conditions listed below occur, THEN GO TO 0POP05-EO-ES11, SI TERMINATION, Step 1:

a. RCS subcooling based on core exit T/Cs - GREATER THAN 35°F [45°F]
b. Total AFW flow to intact SGs - GREATER THAN 576 GPM OR NR level in at least one intact SG - GREATER THAN 14% [34%]
c. RCS pressure GREATER THAN 1745 PSIG AND STABLE OR RISING
d. Pressurizer level - GREATER THAN 8% [44%]

SI REINITIATION CRITERIA

IF EITHER condition listed below occurs, THEN manually START SI pump(s).

o RCS subcooling based on core exit T/Cs - LESS THAN 35°F [45°F]
o Pressurizer level - LESS THAN 8% [44%]

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 ENSURE HHSI pumps running, AND 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 AFWST per 0POP02-AF-0001, AUXILIARY FEEDWATER to prevent inventory problems during cooldown.

PRESSURIZER PORV ISOLATION CRITERIA

IF BOTH of the following conditions occur after completion of Step 10, THEN ISOLATE the PORV by PERFORMING the actions of Step 10:

o ANY PRZR PORV opens on a high PRZR pressure signal
o PRZR pressure is below the PRZR PORV setpoint

SEQUENCER LOADING VERIFICATION

IF a LOOP occurs, THEN PERFORM Addendum 4, Sequencer Loading Verification.

RCP TRIP CRITERIA

IF BOTH conditions occur, THEN TRIP ALL RCPs:
o HHSI pumps - AT LEAST ONE RUNNING
o RCS pressure - LESS THAN 1430 PSIG

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.

LHSI PUMP RESTART CRITERIA

IF RCS pressure lowers in an uncontrolled manner to LESS THAN 415 PSIG, THEN START the LHSI pumps to supply water to the RCS.

RWST CONSERVATION CRITERIA

IF all three CS pumps are injecting, THEN secure one containment spray pump.

Exam Bank No.: 2219 **Last used on an NRC exam:** Never

RO Sequence Number: 6

The following Unit 1 conditions exist:

- MODE 5, cooling down to Mode 6
- RHR trains A and B are is service in full cooling mode (RHR heat exchanger outlet valve is full open).
- RHR train C is in standby.

A loss of offsite power subsequently occurs.

Without operator action, which of the following describes the final status of RHR?

A. No RHR Pumps are running.

B. Only RHR trains A and B are in service in the full cooling mode.

C. RHR trains A and B are in service in the full cooling mode and RHR pump C is running in recirculation mode.

D. All three RHR trains are in service in the full cooling mode.

Answer: A No RHR Pumps are running.

Exam Bank No.: 2219

K/A Catalog Number: APE 025 AK1.01 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.9 **10CFR Reference:** 55.41(b)(7)

Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation

STP Lesson: LOT 201.41 **Objective Number:** 45253

List the equipment that starts on a Mode I, II, and III ASF Load Sequence signal.

Reference: LOT201.41, ESF Load Sequencer, handout pages 18 and 21

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: The RHR pumps will strip off because of the LOOP and will not receive an auto start signal from the sequencer.

B: INCORRECT: Credible because safety related equipment does not have undervoltage trip relays, so once the bus is re-energized it might be reasonable to expect the pumps to restart (the pumps do receive a strip signal from the sequencer though).

C: INCORRECT: Credible because the sequencer does send a signal to the RHR pumps (a strip signal), however if the applicant believed it was a start signal, then all 3 pumps could be running with two cooling (as they were prior to the event) and one on recirc.

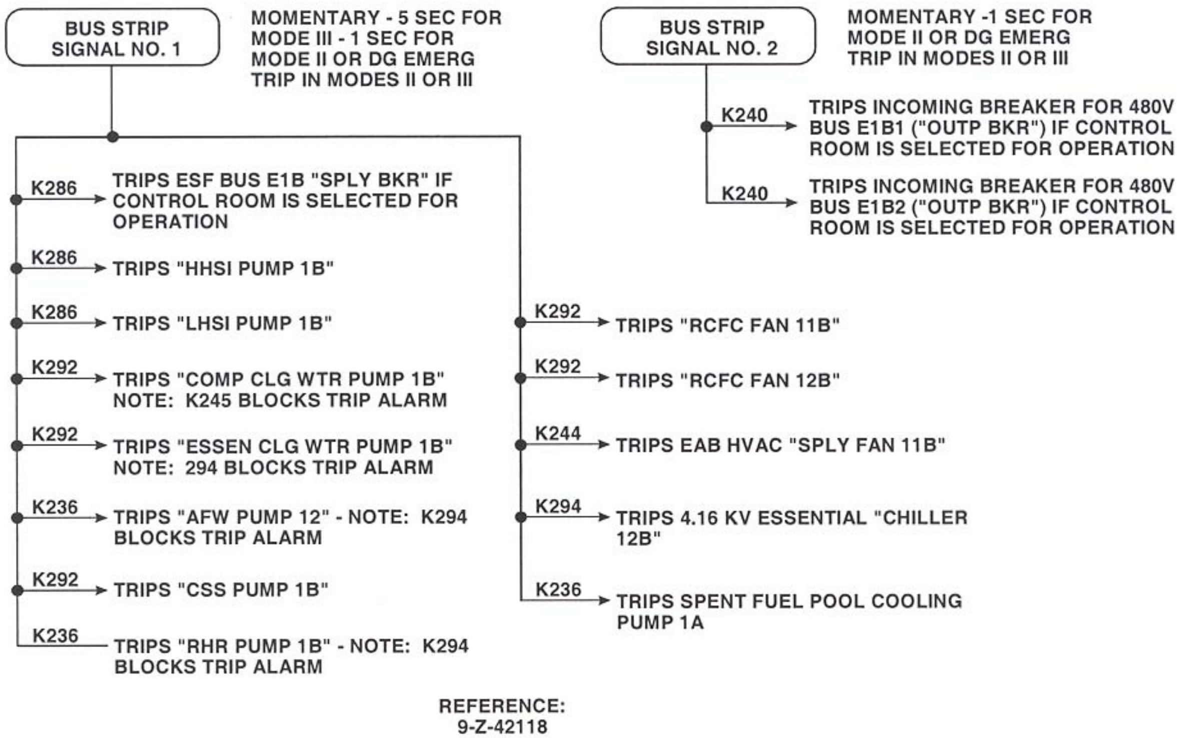
D: INCORRECT: Credible because the sequencer does send a signal to the RHR pumps (a strip signal), however if the applicant believed it was a start signal, then like other safety related equipment it might be running in its safety configuration (which for RHR would be full cooling mode).

Question Level: H **Question Difficulty** 3

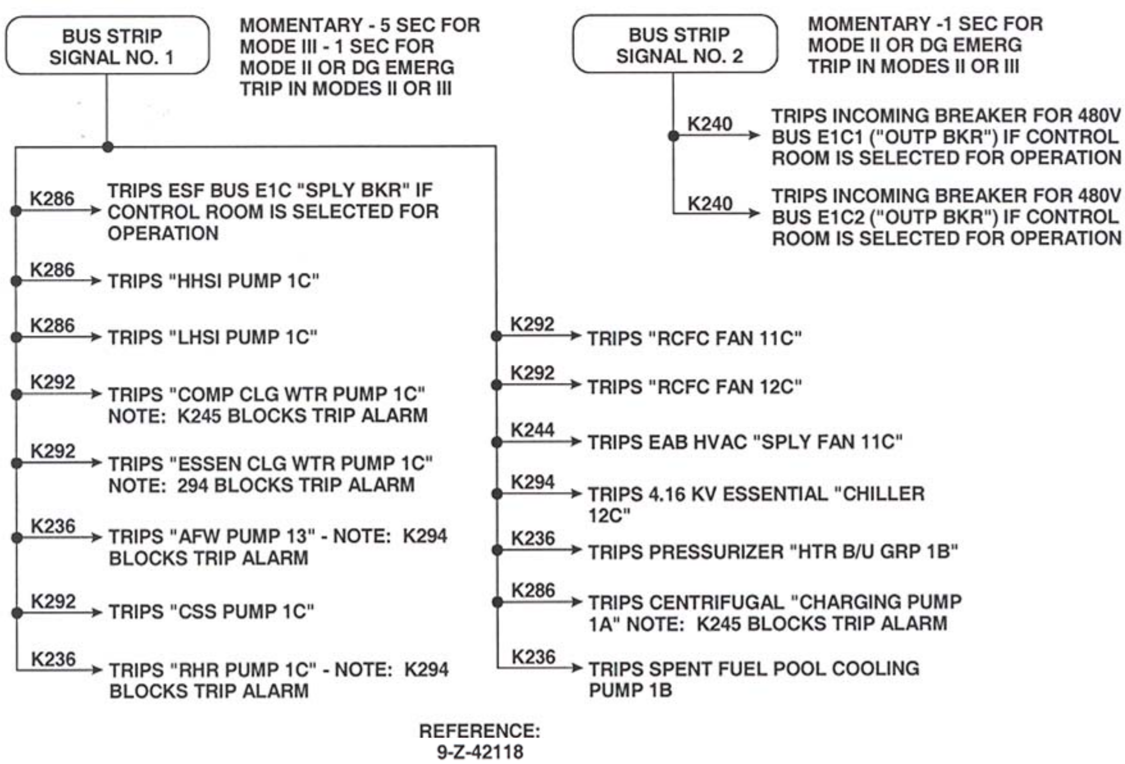
Justification:

The applicant must analyze the given conditions and using knowledge of the RHR and sequencer systems, determine the final status of the RHR pumps.

Train B Bus Strip Signals



Train C Bus Strip Signals



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

Exam Bank No.: 381 **Last used on an NRC exam:** Never

RO Sequence Number: 7

Which of the following is the BASIS for depressurizing intact Steam Generators to 355 psig at the maximum controllable rate during performance of 0POP05-EO-EC00, Loss of All AC Power?

A. To ensure that a heat sink is maintained due to loss of control of the SG PORVs.

B. To maximize Operator control of secondary pressure.

C. To minimize RCS inventory loss through the RCP seals.

D. To prevent challenging the pressurizer safety valves.

Answer: C To minimize RCS inventory loss through the RCP seals

Exam Bank No.: 381

K/A Catalog Number: EPE 055 EK3.02 **Tier:** 1 **Group/Category:** 1

RO Importance: 4.3 **10CFR Reference:** 55.41(b)(5)

Knowledge of the reasons for the following as they apply to the Station Blackout: Actions contained in EOP for loss of offsite and onsite power

STP Lesson: LOT 504.22 **Objective Number:** 82073

Given a copy of a step caution or note from 0POP05-EO-EC00, STATE/IDENTIFY its basis, its purpose and the result of a failure to comply with its requirements.

Reference: WOG ERG EC-0.0, R2, page 118

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: Incorrect - Credible because although the hydraulic pump needs AC power and will not be working, control from the control room is still possible until accumulator pressure is used up.

B: Incorrect - Credible since maximizing operator control of secondary pressure is desirable in a loss of all AC power, however it is the basis for maintaining a faulted or ruptured SG isolated.

C: Correct - To minimize RCS inventory loss through the RCP seals

D: Incorrect - Credible if the PZR PORVs relied on AC power to operate, however the PZR PORVs will still be available to prevent challenging the pressurizer safeties since they are pressure actuated and DC powered.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must recall the basis for the rapid depressurization of the RCS during a loss of all AC.

STEP DESCRIPTION TABLE FOR ECA-0.0 Step 16 – NOTE 1

NOTE: The SGs should be depressurized at maximum rate to minimize RCS inventory loss.

PURPOSE: To inform the operator of the desired rate for depressurization of steam generators

BASIS:

The intact steam generators should be depressurized as quickly as possible, to minimize RCS inventory loss, but within the constraint of controllability. Controllability is required to ensure that steam generator pressures do not undershoot the specified limit.

For the reference plant, the operator can control the secondary depressurization from the control room. In this case, maximum rate means steam generator PORVs full open. For plants that must control the secondary depressurization by local actions, maximum rate must be determined by the control room and local operators based on plant conditions and available communications. A slower rate is acceptable for locally controlled secondary depressurization. See Subsection 2.3.

ACTIONS:

N/A

INSTRUMENTATION:

N/A

CONTROL/EQUIPMENT:

N/A

KNOWLEDGE:

SG depressurization should proceed as quickly as possible and should not be limited by the Technical Specification RCS cooldown limit of 100°F/hr.

PLANT-SPECIFIC INFORMATION:

N/A

Exam Bank No.: 524 **Last used on an NRC exam:** 1995

RO Sequence Number: 8

A reactor trip has occurred on Unit 2.

During the performance of the Immediate Operator actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection, the Reactor Operator notes that the Reactor Trip Breakers are both open, but 4 Control Rods are indicating 18 steps withdrawn.

The crew transitions to 0POP05-EO-ES01, Reactor Trip Response.

Which of the following describes the minimum amount of Boric Acid in gallons that must be added to the RCS and the BASIS for the Boric Acid addition?

Emergency Borate…

A. 3760 gallons or until RCS boron concentration is determined to be greater than 2800 ppm to lower upper range flux to less than 5%.

B. 3760 gallons or until RCS boron concentration is determined to be greater than 2800 ppm to account for the reactivity worth of the stuck rods.

C. 14400 gallons or until RCS boron concentration is determined to be greater than 2800 ppm to lower upper range flux to less than 5%.

D. 14400 gallons or until RCS boron concentration is determined to be greater than 2800 ppm to account for the reactivity worth of the stuck rods.

Answer: B 3760 gallons or until RCS boron concentration is determined to be greater than 2800 ppm to account for the reactivity worth of the stuck rods.

Exam Bank No.: 524

K/A Catalog Number: APE 024 AK3.01 **Tier:** 1 **Group/Category:** 2

RO Importance: 4.1 **10CFR Reference:** 55.41(b)()

Knowledge of the reasons for the following responses as they apply to the Emergency Boration: When emergency boration is required

STP Lesson: LOT 504.06 **Objective Number:** 81674

Given a step, note, or caution from 0POP05-EO-ES01, STATE/IDENTIFY the basis for the step, note or caution and the basis for the action to include the action itself, its purpose and result

Reference: 0POP05-EO-ES01, step 4

Attached Reference ☑ **Attachment:** 0POP05-ES-ES01rev.26, page 5

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible because adding acid would lower power, however if power was actually >5%, a different procedure would be in use.

B: CORRECT - 940 gallons per rod of emergency boration is required for each rod stuck at or below 18 steps (940x4=3760). The basis is to account for stuck rod reactivity worth and ensure adequate shutdown margin.

C: INCORRECT - Credible since this is the amount of boration that would be required if the applicant did not read the procedure closely enough and used the per rod value of boric acid when the rod is more than 18 steps withdrawn (18 steps is the transition point). The basis is credible because adding acid would lower power, however if power was actually >5%, a different procedure would be in use.

D: INCORRECT - Credible since this is the amount of boration that would be required if the applicant did not read the procedure closely enough and used the per rod value of boric acid when the rod is more than 18 steps withdrawn (18 steps is the transition point).

Question Level: H **Question Difficulty** 3

Justification:

The applicant must determine the correct amount of boric acid to add from the given conditions and recall the basis for the action.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___4	**VERIFY Control Rods Fully Inserted** o All rod bottom lights - LIT	PERFORM the following: a. IF two OR more control rod bottom lights NOT lit, THEN: 1) Emergency BORATE 940 GALLONS of boric acid (60 ppm) for each control rod 18 steps OR LESS. _____ X 940 gals = _______ gals # of rods 2) Emergency BORATE 3600 GALLONS of boric acid (228 ppm) for each control rod GREATER THAN 18 steps. ____ X 3600 gals = _______ gals # of rods OR 3) Emergency BORATE until RCS Cb GREATER THAN 2800 PPM ppm. b. IF DRPI has failed, THEN: 1) INITIATE emergency boration. 2) WHEN DRPI has been restored OR RCS Cb GREATER THAN 2800 PPM ppm, THEN SECURE emergency boration. -------------------------------------

Exam Bank No.: 2160 **Last used on an NRC exam:** Never

RO Sequence Number: 9

Chemistry has reported a rise in reactor Coolant activity and the Unit Supervisor has entered 0POP04-RC-0001, High Reactor Coolant System Activity.

Which of the following correctly describes an action that should be taken and the reason for the action?

A. Start a second Centrifugal Charging Pump and place all letdown orifices in service to maximize purification flow through the in service Mixed Bed Demineralizer.

B. Start a second Centrifugal Charging Pump and place all letdown orifices in service to maximize filtration by the reactor coolant filter.

C. Place a Cation Bed Demineralizer in service to remove lithium which reduces RCS pH and minimizes the chances of an RCS crud burst.

D. Place a Cation Bed Demineralizer in service to maximize effective purification.

Answer: D Place a Cation Bed Demineralizer in service to maximize effective purification.

Exam Bank No.: 2160

K/A Catalog Number: APE 076 AK3.06 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.2 **10CFR Reference:** 55.41(b)(5)

Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Actions contained in EOP for high reactor coolant activity.

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: POP04-RC-0001 step 7 basis

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because raising letdown flow is a procedural action and doing so will raise deminerilizer flow and help lower activity. However, starting and running a second charging pump would be required with all orifices in service and would be a procedure violation.

B: INCORRECT: Credible because raising letdown flow is a procedural action and doing so will raise flow through the filter and help lower activity. However, starting and running a second charging pump would be required with all orifices in service and would be a procedure violation.

C: INCORRECT: Credible because the cation demineralizer is used during normal operation to lower lithium concentration in the RCS, however a procedural caution warns that doing so may initiate a crud burst (not minimize).

D: CORRECT: Placing a cation bed in service will aid in removing fission products.

Question Level: F **Question Difficulty** 3

Justification:

The applicant musy have knowledge of procedural requirements and basis.

STEP DESCRIPTION FOR 0POP04-RC-0001 STEP 7.0

STEP: CHECK Cation Demineralizers In Service

PURPOSE: To inform the operator to place cation demineralizers in service to facilitate reducing RCS activity if not already in service in Step 4.0.

BASIS: Adequate resin bed capacity will ensure maximum effective purification.

ACTIONS: Place cation demineralizers in service.

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: N/A

KNOWLEDGE: N/A

Exam Bank No.: 1296 **Last used on an NRC exam:** 2005

RO Sequence Number: 10

Unit 2 is operating in Mode 1 with the following Component Cooling Water (CCW) Pump lineup:

- CCW Pump 2A – running
- CCW Pump 2B – standby
- CCW Pump 2C – tagged out for maintenance

Subsequently:

- A failure of CCW Pump 2A discharge valve causes it to drift partially closed.
- CCW Header pressure drops to 85 psig.
- Normal letdown has remained in service.

Which of the following describes the system response an operator should expect to observe?

A. Letdown flow diverts to the Recycle Holdup Tank (RHUT)

B. Initial rise then return to normal in letdown temperature downstream of the Letdown Heat Exchanger

C. Initial rise then return to normal in seal water return temperature downstream of the Seal Water Heat Exchanger

D. Letdown flow diverts to the Reactor Coolant Drain Tank (RCDT)

Answer: B Initial rise then return to normal in letdown temperature downstream of the Letdown Heat Exchanger

Exam Bank No.: 1296

K/A Catalog Number: APE 026 AA1.06 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(7)

Ability to operate and/or monitor the following as they apply to the Loss of Component Cooling Water: Control of flow rates to components cooled by the CCWS

STP Lesson: LOT 201.12 **Objective Number:** 5213

Given a plant or system condition, PREDICT the operation of the Component Cooling Water System.

Reference: LOT201.06.HO.01, pages 18 and 29; LOT201.12.HO.01, page 15

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: This distractor is credible because letdown flow will divert to the RHUT but only on high VCT level not high letdown temperature caused by reduced CCW flow and pressure.

B: CORRECT: Lowering CCW flow will cause letdown temp to rise, then as letdown heat exchanget TCV-4494 opens, the temperature will return to normal. NOTE: CCW Pumps would not change status because a header pressure of 85 psig would not be low enough to start a standby CCW pump.

C: INCORRECT: This distractor is credible because the Seal Water Heat Exchanger is cooled by CCW but a TCV does not control CCW flow to the Seal Water Heat Exchanger so in this situation, temperature will rise and remain at the higher temperature.

D: INCORRECT: This distractor is credible because CVCS letdown can be diverted to the RCDT but only if Excess Letdown is in service.

Question Level: H **Question Difficulty** 3

Justification:

Requires ability to determine the effect the reduced system flow will have on heat transfer in the CVCS letdown heat exchanger and the response of the downstream temperature control valve.

to fail open on loss of air or electrical power; however, if it were to fail shut, a manual bypass valve is provided in order to continue letdown operation. TCV-381A, the return from the BTRS Reheat Heat Exchanger, fails closed on loss of air or electrical power.

The reactor coolant purification pump is used to circulate water from the RHR system during cold or refueling shutdown for purification. The purification pump capacity of 450 gpm provides maximum purification during shutdown low pressure operations. The purification pump is a centrifugal pump designed for 600 psig and 400°F.

During this mode of purification, both mixed beds and the RHR return path downstream of the reactor coolant filter is used. The purification pump has "STOP, NORMAL, START" control switch on CP-004 with a P-T-L position.

Over pressure protection is provided by an alarm which actuates on a R.C. purification pump discharge pressure of 540 psig and by a relief valve which relieves to the VCT at 600 psig. The purification pump is located on the 10 foot elevation of the Mechanical Auxiliary Building (MAB).

The R.C. Purification Pump Discharge Valve (HCV-133) is used to control the pump flow rate. The hand controller is located on CP-004.

The letdown heat exchanger (Figure 5) is U-tube type heat exchanger with letdown through the tubes being cooled by component cooling water on the shell side. The component cooling water control valve automatically controls the temperature of the letdown flow from the letdown heat exchanger at 115°F. If the letdown outlet temperature increases above 120°F, a letdown high temperature alarm will sound on the main control board. The letdown heat exchanger is located on the 10 foot elevation of the MAB.

Low Pressure Letdown Control Valve (PCV-135) maintains the pressure downstream of the letdown orifices automatically at ~350 psig. Maintaining this backpressure prevents the letdown flow from "flashing to steam" as it passes through the orifice valves. Flashing would cause excessive erosion of the orifices. During solid plant operations,, the low pressure letdown control valve's function is to maintain the pressure of the RCS. PCV-135 fails open on a loss of control air to the valve, or a loss of control voltage to the air controller. A manually operated bypass valve is installed around PCV-135 to allow continued operation with normal letdown flow if the valve should stick or fail. A letdown high pressure alarm at 500 psig is provided to warn the operator. The controller for PCV-135 is located on CP-004.

Low pressure Letdown Line Relief Valve will relieve to the VCT when the pressure in the low pressure letdown line increases to 300 psig.

Letdown Flow is read on FI-132 on CP-004. Letdown flow is sensed by FT-132 downstream of PCV-135. This instrument actuates a letdown Hi/Lo flow alarm at 260 gpm/70 gpm respectively.

Letdown flow is continuously sampled by a liquid Radiation Monitor (RIT-8039) downstream of PCV-135. The differential pressure across the orifice used to measure

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.

The pump net positive suction head requirement for all modes of plant operation is satisfied by locating the pump lower than the surge tank. The surge tank centerline is at elevation 68' 6" while the pump centerline is at elevation 13' 9-1/4". The CCW pump motors are powered from the 4.16 KV Class 1E power distribution system.

4.4.1 For normal operation, the CCW trains are selected to be off, running or in standby for automatic start, along with the respective ECW train, by CCW/ECW Mode Selector Switches located on CP-002. Normally one train is in "OFF", one is in "Standby" and one is in "RUN".

A. When selected for "Standby", low pressure in the CCW common header (76 psig) or a low pressure in the other two ECW loops (30 psig) initiates automatic startup of the pump after a 15 second delay (if LOOP or SI does not exist). The delay in auto starting allows for switching the operating pumps during normal operations. Automatic startup of the corresponding ECW pump will also occur simultaneously.

B. When selected to "Run", the CCW and ECW pump for that train will start unless a LOOP or SI signal is present.

C. If two of the three selector switches are in "Off", a "Standby Train Not Selected" alarm will annunciate.

4.4.2 Transfer of control and indication for the CCW pumps from the transfer switch panels to the main control room is provided by "Local/Remote" switches located on the transfer switch panels in the switchgear rooms.

A. When in "local", a BYPASS INOP alarm is sounded and all automatic functions are disabled.

4.4.3 Individual STOP/AUTO/START control switches are provided on CP-002 and the transfer switch panels along with running (red) and stopped (green) pump status lights. The control room switch also has a Pull-To-Lock (PTL) feature that will stop the pump under all conditions.

A. The status lights indicate only at the location selected by the Local/Remote switch.

4.4.4 Auto starts

Exam Bank No.: 1577 **Last used on an NRC exam:** 2010

RO Sequence Number: 11

An operator action of 0POP05-EO-FRS1, Response to Nuclear Power Generation – ATWS, is to "Ensure 480V LC 1K1 (2K1) and 1L1 (2L1) feeder breakers open".

This step will de-energize power to the....

A. Rod Drive MG Set motors. Opening only one of the breakers will cause a reactor trip.

B. Rod Drive MG Set motors. Both breakers must be opened to cause a reactor trip.

C. Reactor Trip Breaker shunt trip coils. Opening only one of the breakers will cause a reactor trip.

D. Reactor Trip Breaker shunt trip coils. Both breakers must be opened to cause a reactor trip.

Answer: B Rod Drive MG Set motors. Both breakers must be opened to cause a reactor trip.

Exam Bank No.: 1577

K/A Catalog Number: EPE 029 EK2.06 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(7)

Knowledge of the interrrelations between components following an ATWS: Breakers, relays, and disconnects

STP Lesson: LOT 201.18 **Objective Number:** 3069

IDENTIFY major components, system interfaces, interlocks and relative location of components and instrumentation by drawing and labeling a block diagram of the Rod Control System.

Reference: LOT 201.18 PowerPoint slide 19

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because only one trip breaker must be open to trip the reactor, but both MG sets must be de-energized to initiate a reactor trip.

B: CORRECT: The Load Centers supply power to the MG motors. Either MG set providing power to the Rod Control System will be sufficient to power the rod drives (design redundancy) so both MG must be de-energized as stated.

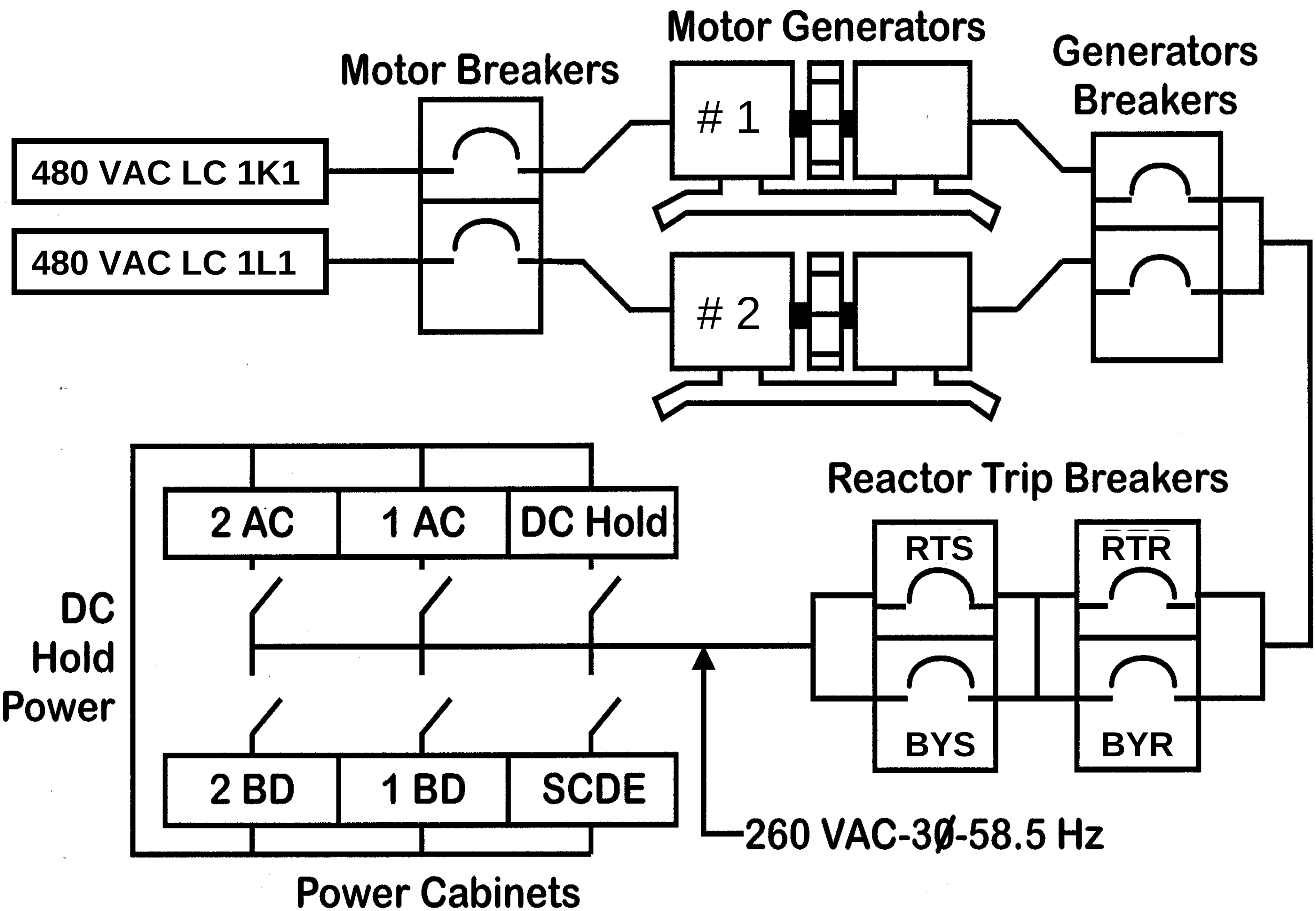
C: INCORRECT: Credible because the trip breakers will open if the shunt trip coil is de-energized, however it is powered from a different source. Part two is credible because only one trip breaker must be open to trip the reactor, but both MG sets must be de-energized to initiate a reactor trip.

D: INCORRECT: Credible because the trip breakers will open if the shunt trip coil is de-energized, however it is powered from a different source.

Question Level: F **Question Difficulty** 3

Justification:

Must know the distribution for rod drive power including the design redundancy that must be accounted for to perform a reactor trip.



Motor Generators
Motor Breakers
Generators Breakers
480 VAC LC 1K1
480 VAC LC 1L1
1
2
Reactor Trip Breakers
2 AC
1 AC
DC Hold
RTS
RTR
BYS
BYR
DC Hold Power
2 BD
1 BD
SCDE
260 VAC-3Ø-58.5 Hz
Power Cabinets

Exam Bank No.: 1651 **Last used on an NRC exam:** 2007

RO Sequence Number: 12

A liquid release is in progress from a Waste Monitor Tank (WMT).

Which of the following correctly describes operation of the system if the Liquid Release Rad Monitor, RT-8038 reaches its HIGH alarm setpoint?

Liquid Waste Discharge Valve, FV-4077, should…

A. CLOSE to stop flow from the WMT. If the valve fails to close, the Control Room operator must manually close the valve from the RM-11 console.

B. RE-POSITION to recirc the contents of the WMT. If the valve fails to re-position, the Control Room operator must manually re-position the valve to recirc from the RM-11 console.

C. CLOSE to stop flow from the WMT. If the valve fails to close, a Plant Operator will have to be dispatched to close the valve using handswitch on Rad Waste Controlroom panel.

D. RE-POSITION to recirc the contents of the WMT. If the valve fails to re-position, a Plant Operator will have to be dispatched to re-position the valve using handswitch on Rad Waste Controlroom panel.

Answer: D Liquid Waste Discharge Valve, FV4077, will re-position to recirc the contents of the WMT. If the valve fails to re-position, a Plant Operator will have to be dispatched to re-position the valve using handswitch on Rad Waste Controlroom panel.

Exam Bank No.: 1651

K/A Catalog Number: APE 059 AA1.01 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(13)

Ability to operate and/or monitor the following as they apply to the Accidental Liquid Radwaste Release: Radioactive-liquid monitor

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: A. Boron Recycle System, B. Gaseous Waste Processing System, C. Liquid Waste, Processing System, D. Turbine Generator Building Sump and Drain System, E. Condensate Polishing System, F. Steam Generator Blowdown System, G. Containment Building, H. Electrical Auxiliary Building and Control Room Envelope HVAC, I. Fuel Handling Building Ventilation System

Reference: LOT202.41 student handout page 26

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Valve position is credible because the desire would be to stop the discharge (which closing would). Valve control is credible since the monitor is displayed on the RM-11 console in the CR and has some control functions but only for the monitor, the valve cannot be operated from the panel.

B: INCORRECT - Credible because the monitor is displayed on the RM-11 console in the CR and has some control functions but only for the monitor, the valve cannot be operated from the panel.

C: INCORRECT - Credible because the desire would be to stop the discharge (which closing would).

D: CORRECT- FV4077 repositions to recirc tank if it fails to actuate a plant operater will have to go to local hand switch.

Question Level: H **Question Difficulty** 3

Justification:

The applicant requires a knowledge of the automatic actions associated with the affected radiation monitor and where the valve can be operated from.

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.: 1733 **Last used on an NRC exam:** 2007

RO Sequence Number: 13

Unit 1 is at 15% reactor power during a plant shutdown.

The Main Generator is at 210 MWe.

A large transient on the grid causes switchyard frequency to lower to 56.5 Hz.

Assuming no operator action, which of the following correctly identifies the status of the reactor coolant pump breakers and reactor trip breakers following the grid transient?

	Reactor Coolant Pump Breakers	Reactor Trip Breakers
A.	Open	Open
B.	Open	Closed
C.	Closed	Closed
D.	Closed	Open

Answer: A Open, Open

Exam Bank No.: 1733

K/A Catalog Number: EPE 007 EK2.02 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(7)

Knowledge of the interrelations between a reactor trip and the following: Breakers, relays and disconnects.

STP Lesson: LOT 201.20 **Objective Number:** 26026

Given a description of plant conditions DETERMINE if an automatic reactor trip signal would be generated.

Reference: LOT201.20, Handout page 41

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT - The reactor trip setpoint for underfrequency is 57.2 Hz. If reactor power is greater than P-7, then the underfrequency trip will open the reactor trip breakers and the reactor coolant pump breakers. The same signal causes both sets of breakers to trip.

B: INCORRECT - This distractor is credible because the RX Trip breakers would not open if Reactor is less than 10% power.

C: INCORRECT - This distractor is credible if the applicant believes the frequency did not lower enough to cause a trip of the reactor or RCPs.

D: INCORRECT - This distractor is credible if the applicant is aware of the underfrequency reactor trip, but did not realize the RCPs also receive a trip signal (they do not trip on undervoltage, althought the reactor still trips).

Question Level: H **Question Difficulty** 3

Justification:

From the given conditions, the applicant must use their knowledge of reactor trip setpoints (underfrequency), permissives (P-7) and interlocks (tripping of all RCPs) to determine the correct response.

4. Time delays are incorporated to prevent spurious trips from momentary electrical transients

   a. Delay is set at $\leq$ .4 Seconds from trip of two or more RCP Bus circuit breakers to signal reaching reactor trip breakers. Tech Specs require $\leq$ 1.2 seconds.

5. Alarms and annunciators

   a. Located on CP-005

   b. RCP Bus Undervolt

E. Reactor coolant pump (TP .42)

1. Provide reactor core protection against DNB as a result of underfrequency in more than one RCP

   a. Loss of forced reactor coolant flow

2. $\leq$ 57.2 HZ; 2/4 UF sensors and not blocked (i.e. 2/4 Pumps having UF condition)

3. Automatically blocked below P-7

4. Will also trip open all RCP breakers to <u>allow for coastdown</u>

5. Time delays are incorporated to prevent spurious trips from momentary electrical transients

   a. Delay is set at $\leq$ 0.6 Seconds from time underfrequency trip setpoint is reached to signal reaching reactor trip breakers

6. The signal to trip open all RCP breakers is blocked if the RCP Undervoltage relay is actuated.

7. Alarms and annunciators

   a. Located on CP-005

   b. RCP Bus Underfrequency RX Trip

Exam Bank No.: 1820 **Last used on an NRC exam:** 2011

RO Sequence Number: 14

Given the following:

- Unit 1 is operating at full power.
- 480V LC E1A2 TRBL alarm occurs.
- 480V Load Center E1A2 Bus Volts = 0 volts
- Annunciator 125 VDC SYSTEM E1A11 TRBL alarms.
- Channel 1 BATT CUR indicates 30 amps discharge.

Assuming the plant responds as designed and without operator action, …

A. DP001 is now powered from its Voltage Regulating Transformer.

B. DP1201 is now powered from its Voltage Regulating Transformer.

C. Bus E1A11 is being powered from its respective ESF Battery.

D. Bus E1A11 is being powered from its Standby Battery Charger.

Answer: C Bus E1A11 is being powered from its respective ESF Battery.

Exam Bank No.: 1820

K/A Catalog Number: APE 058 AA1.03 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.1 **10CFR Reference:** 55.41(b)(7)

Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Vital and battery bus components.

STP Lesson: LOT 201.37 **Objective Number:** 63901

GIVEN a loss of power, PREDICT the operation of the class 1E DC Electrical Distribution System to include automatic actions and interlocks.

Reference: LOT 201.37 PowerPoint slide 14, LOT201.38 PowerPoint slide 93

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because the 480v inputs to the 120v DP panel systems are split between the two load centers and the applicant may believe the loss of the 480v input will cause a loss of the normal supply, causing it to swap to the voltage regulating transformer. However, as long as battery voltage is available, the swap to the voltage regulating transformer will not occur.

B: INCORRECT: Credible because the 480v inputs to the 120v DP panel systems are split between the two load centers and the applicant may believe the loss of the 480v input will cause a loss of the normal supply, causing it to swap to the voltage regulating transformer. However, as long as battery voltage is available, the swap to the voltage regulating transformer will not occur.

C: CORRECT: The Class 1E 125 VDC Bus E1A11 would not lose power with the given conditions. The symptoms indicate that a loss of the in-service Battery Charger has occurred. This will not result in a loss of power to Class 1E 120 volt vital DP 1201 and 001. Class 1E Battery E1A11 will automatically supply power to DP 1201 and 001 through their respective inverters.

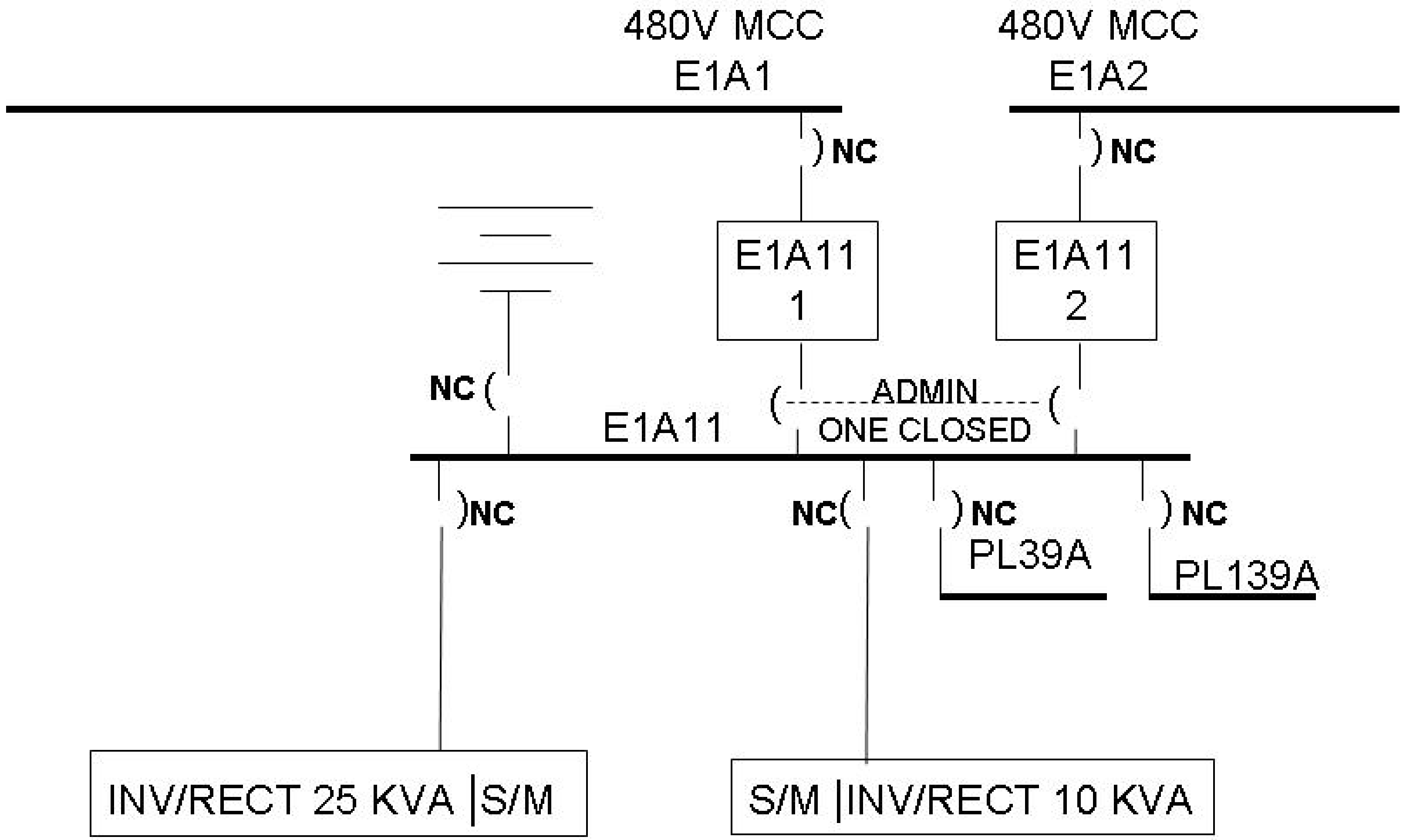
D: INCORRECT: Credible because each battery has two chargers, each powered from a different load center, but only one is normally in service.

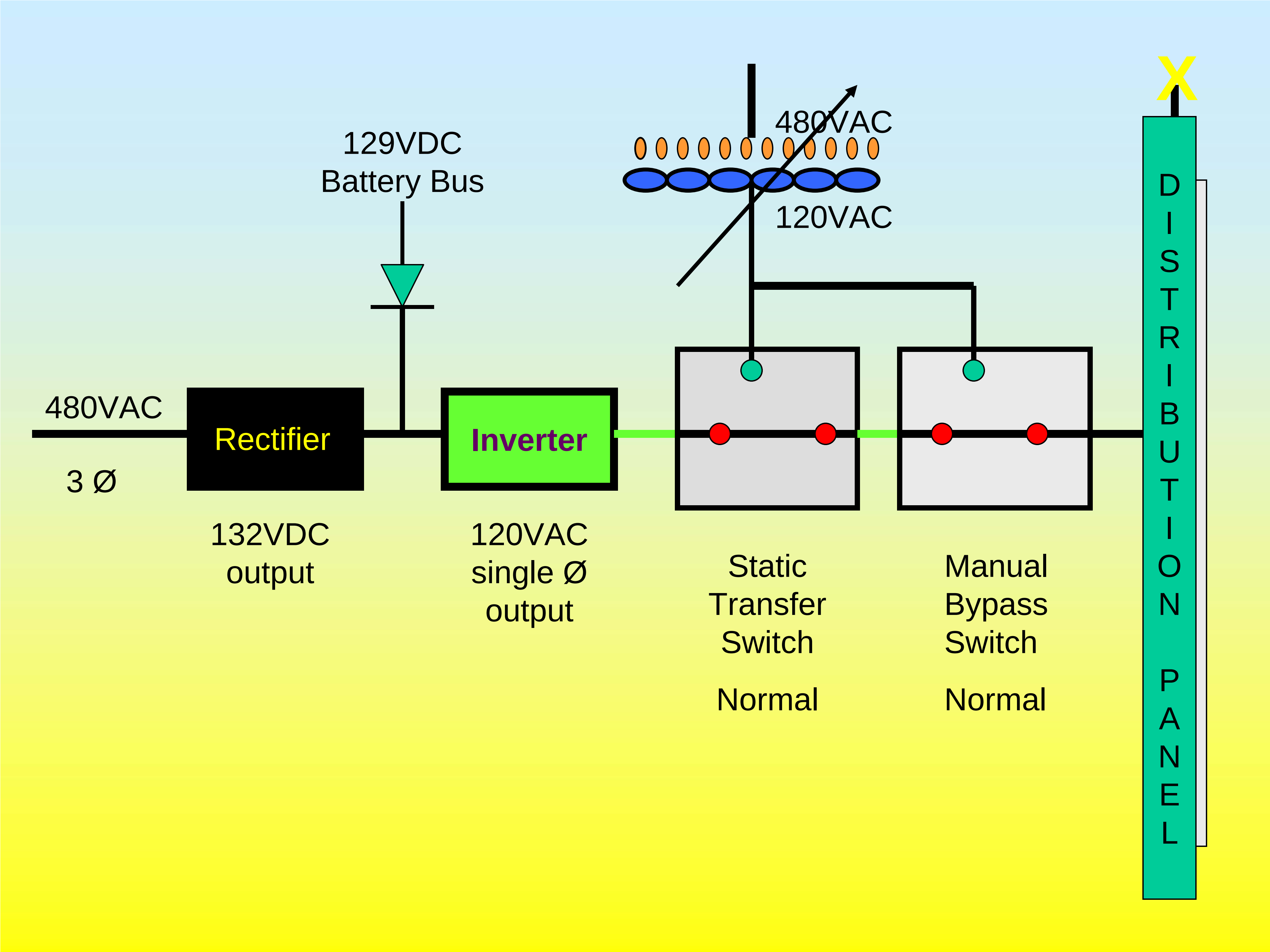
Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine whether a loss of Vital DC power has occurred from the symptoms given. Then, based on what was lost, determine how the 125 VDC System will respond.

125 VDC Train A Channel I





X
129VDC
Battery Bus
480VAC
120VAC
480VAC
3 Ø
Rectifier
Inverter
132VDC
output
120VAC
single Ø
output
Static
Transfer
Switch
Normal
Manual
Bypass
Switch
Normal
DISTRIBUTION PANEL

Exam Bank No.: 1863 **Last used on an NRC exam:** 2009

RO Sequence Number: 15

Given the following conditions:

- Unit 1 is raising Reactor Power and is currently at 40% power.

Subsequently:

- Instrument Air pressure began to lower and is currently at 95 psig and trending down slowly.
- The Control Room Crew is working through 0POP04-IA-0001, Loss of Instrument Air.

Based on current given conditions, which of the following describes the next appropriate crew response in accordance with 0POP04-IA-0001, Loss of Instrument Air?

A. Isolate CVCS Charging and Letdown flow.

B. Trip the Main Turbine and Isolate Main Steam.

C. Trip the Reactor and Ensure the Main Turbine is tripped.

D. Verify that the Instrument Air to Service Air Isolation Valve has closed.

Answer: D Verify that the Instrument Air to Service Air Isolation Valve has closed

Exam Bank No.: 1863

K/A Catalog Number: APE 065 G2.4.11 **Tier:** 1 **Group/Category:** 1

RO Importance: 4.0 **10CFR Reference:** 55.41(b)(10)

Loss of Instrument Air: Knowledge of abnormal condition procedures.

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 step 6

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: This distractor is credible because the procedure states in a note at the beginning that CVCS letdown isolation valve, FV-0011, will start to drift closed at 80 psig. If the crew lost letdown flow then they would also isolate charging flow.

B: INCORRECT: This distractor is credible because in the basis of the procedure it states that MSIVs will begin to drift closed at 56 psig. With the Reactor at 40% power and less than P-9, if the crew closed the MSIVs then they could trip the main turbine and not the Reactor.

C: INCORRECT: This distractor is credible because on the CIP of the procedure it states that if IA pressure lowers to 60 psig then Trip the Reactor. If the crew tripped the reactor they would have to ensure the Main Turbine tripped as well.

D: CORRECT: This question is answered correctly by having knowledge of the different IA pressures that effect IA valves and require specific actions. In this case with IA pressure at 95 psig and lowering the next appropriate action is to verify IA to SA Isolation valve is closed. The IA to SA isloation valve set point for closing is given in a note at the begining of the procedure and the valve is verified closed in step 6 in the body of the procedure when IA pressure is below 100 psig.

Question Level: F **Question Difficulty** 3

Justification:

Must know the Off Normal procedure requirements`

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 5.0	**CHECK IA Pressure - LESS THAN 100 PSIG**	PERFORM the following: a. IF IA pressure lowers to less than 100 psig, THEN GO TO Step 6.0. b. GO TO Step 14.0.
_____ 6.0	**CHECK IA Pressure - LESS THAN 90 PSIG**	PERFORM the following: a. VERIFY Service Air Flow Secured on ICS Display IA-001 “SA Flow {CFM}” 1) IF Service Air Flow is NOT Isolated, THEN DISPATCH an Operator to perform the following: a) VERIFY “Service Air Isolation Valve N1(2)IA-PV-9785” Closed. {29 ft TGB} b) IF N1(2)IA-PV-9785 is NOT closed, THEN CLOSE “1(2)-SA-9982 SERVICE AIR SYSTEM PV-9785 INLET ISOLATION VALVE”.{29 ft TGB Between IA Dryer 11(21) and Dry Receiver 12(22)} b. ENSURE all available IA Compressors running OR aligned to Start/Load. c. IF IA pressure lowers to less than 90 psig, THEN GO TO Step 7.0. d. GO TO Step 14.0.

Exam Bank No.: 1864 **Last used on an NRC exam:** 2009

RO Sequence Number: 16

Given the following:

- Unit 1 operators are establishing RCS bleed and feed in accordance with 0POP05-EO-FRH1, Loss of Secondary Heat Sink.
- While verifying RCS bleed path per step 13, the Reactor Operator observes that ONE of the Pressurizer PORV's will not open.

Which of the following describes the appropriate response in accordance with 0POP05-EO-FRH1, Loss of Secondary Heat Sink, step 13, and the reason for this response?

A. Open the Reactor Vessel Head Vent valves because the RCS may not depressurize sufficiently to permit adequate SI flow to remove core decay heat.

B. Close the open PORV and continue efforts to restore AFW flow because one PORV will not depressurize the RCS sufficiently to allow SI to maintain RCS inventory.

C. No action is required because the RCS will still depressurize sufficiently with one PORV open to permit adequate SI flow to remove core decay heat.

D. Close the open PORV, then open the Reactor Vessel Head Vent valves to restrict the mass loss sufficiently to ensure RCS inventory can be maintained with SI.

Answer: A Open the Reactor Vessel Head Vent valves because the RCS may not depressurize sufficiently to permit adequate SI flow to remove core decay heat.

Exam Bank No.: 1864

K/A Catalog Number: EPE E05 EK2.1 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.7 **10CFR Reference:** 55.41(b)(5)

Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

STP Lesson: LOT 504.33 **Objective Number:** 83085

DESCRIBE the indications and anticipated readings used to determine that the Reactor Coolant System bleed path is adequate

Reference: 0POP05-EO-FRH1 step 13
WOG ERG Background FRH.1 (Rev 2)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT: Reason and action IAW with the references cited.

B: INCORRECT: This distractor is credible because continuing efforts to restore AFW flow and returning to the beginning of the procedure is an action if adequate RCS FEED path cannot be established.

C: INCORRECT: This distractor is credible because it identifies a lack of knowledge with the function of the PZR PORVs and their interrelation with the Loss of Secondary Heat Sink. One PORV will NOT depressurize the RCS sufficiently to allow enough SI flow to maintain inventory and remove decay heat.

D: INCORRECT: This distractor is credible because it identifies a lack of knowledge with the function of the PZR PORVs and the Reactor Head Vent valves and their interrelation with the Loss of Secondary Heat Sink. Opening the Reactor Vessel Head Vent Valves provides additional depressurization of the RCS but is still not enough to equal 2 PZR PORVs being open even though this same action is performed if no PZR PORVs can be opened. If only one PZR PORV can be opened, then it needs to stay open.

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine appropriate procedure response for given plant conditions and understand why those actions are necessary.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___13	**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" d. ALIGN any available low pressure water source to the SG(s). o REFER TO ADDENDUM 2, ESTABLISHING FEED FLOW WITH CONDENSATE SYSTEM. e. <u>IF</u> low pressure water source can <u>NOT</u> be aligned, <u>THEN</u> GO TO Step 14. f. DEPRESSURIZE at least one intact SG to atmospheric pressure using SG PORV to inject low pressure water source. g. <u>IF</u> SG PORVs can <u>NOT</u> be opened from the control room, <u>THEN</u>: 1) ENSURE SG PORV in manual. 2) DEPRESS and HOLD SG PORV(s) down arrow pushbutton for GREATER THAN 20 SECONDS. 3) DISPATCH operator to open one SG PORV per ADDENDUM 3, SG PORV LOCAL OPERATION. -------------------------------------
___14	**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. -------------------------------------

Exam Bank No.: 2150 **Last used on an NRC exam:** Never

RO Sequence Number: 17

Given the following:

- Unit 1 is operating at 100% power
- All control systems are operating in automatic
- One Pressurizer Power Operated Relief Valve (PORV) fails full open
- No Operator actions are taken

Considering the leak rate through the open Pressurizer PORV, which of the following describes the plant response to this event?

A. An Over-Temperature Delta-T reactor trip will occur along with a low Pressurizer pressure Safety Injection. RCS pressure will stabilize above the shutoff head for the LHSI Pumps due to injection by the HHSI Pumps.

B. An Over-Temperature Delta-T reactor trip will occur along with a low Pressurizer pressure Safety Injection. RCS pressure will stabilize below the shutoff head of the LHSI Pumps.

C. An Over-Power Delta-T reactor trip will occur along with a low Pressurizer pressure Safety Injection. RCS pressure will stabilize above the shutoff head for the LHSI Pumps due to injection by the HHSI Pumps.

D. An Over-Power Delta-T reactor trip will occur along with a low Pressurizer pressure Safety Injection. RCS pressure will stabilize below the shutoff head of the LHSI Pumps.

Answer: A An Overtemperature Delta-T reactor trip will occur along with a low Pressurizer pressure Safety Injection. RCS pressure will stabilize above the shutoff head for the LHSI Pumps due to injection by the HHSI Pumps.

Exam Bank No.: 2150

K/A Catalog Number: APE 008 AA2.25 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.8 **10CFR Reference:** 55.41(b)(5)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Expected leak rate from open PORV or code safety

STP Lesson: LOT 501.21 **Objective Number:** 501215

Given a set of 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: LOT501.21, handout page 8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT - An OTDT trip will occur first at this power level in response to a lowering pressure. RCS pressure will continue to drop until it stabilizes based on HHSI pump flow (SBLOCA response)

B: INCORRECT - This distractor is credible because it shows a lack of knowledge of the amount of flow that can come from an open PZR PORV. RCS pressure will lower, but not to the extent of having LHSI pumps injecting based on the leak size.

C: INCORRECT - This distractor is credible because with a lowering pressure at full power, OTDT trip will occur first. Although OPDT has a variable setpoint like OTDT, OPDT is not changed by a lowering PZR Pressure. RCS pressure response is correct.

D: INCORRECT - This distractor is credible because with a lowering pressure at full power, OTDT trip will occur first. Although OPDT has a variable setpoint like OTDT, OPDT is not changed by a lowering PZR Pressure. It also shows a lack of knowledge of the amount of flow that can come from an open PZR PORV. RCS pressure will lower, but not to the extent of having LHSI pumps injecting based on the leak size.

Question Level: H **Question Difficulty** 3

Justification:

Applicant must have an understanding of the leak rate through a PORV, then apply that leak to the RCS to determine plant response.

Inadvertent Opening of a Pzr Safety or Relief Valve

Event Definition

The inadvertent opening of a Pzr safety or relief valve is defined as an accidental depressurization of the RCS caused by the spurious actuation of a Pzr safety or relief valve.

Major Concerns

The major concerns associated with the unmitigated inadvertent opening of a Pzr safety or relief valve is possible fuel cladding damage resulting from the decrease in RCS pressure and subsequent violation of the safety analysis limit DNBR value.

Event Hazards/Challenges

The hazards and challenges associated with the inadvertent opening of a Pzr safety or relief valve are:

- A challenge to the fuel and fuel cladding due to the rapid reduction in the RCS pressure along with the high power and could violate the safety analysis limit DNBR.
- A radiological hazard would be created if the reactor coolant water is discharged to the containment through the PRT.

Analysis Objective

The objective of the analysis is to prove that the reactor protection system will automatically terminate the event prior to DNB occurring.

NOTE: Following reactor trip, the operator is expected to be able to isolate the open valve (assuming the failure is a PORV) and prevent further adverse reactor conditions. If the valve can not be isolated (such as a safety valve), the event is no longer a RCS depressurization but a small-break LOCA. The long term plant response due to the opening of a valve that can not be isolated is bounded by the limiting small-break LOCA.

Types of Accidents Analyzed

The accident analyzed is the inadvertent opening of a Pzr safety valve while at full power operation.

- A Pzr safety valve will relieve approximately twice the steam flow rate of a relief valve, and will therefore allow a much more rapid depressurization upon opening.

Inadvertent Opening of a Pzr Safety or Relief Valve *(continued)*

Sequence of Events

DBD, Module 17, Table 17-4, Figures 17-1 to 17-3

The sequence of events is shown below.

Time (sec)	Pzr Pressure	Core Average Temperature	Nuclear Power	DNBR
0-21	Pzr pressure starts decreasing immediately after the Pzr safety valve opens. • Pressure has decrease enough to cause a reactor trip on OTΔT at 20.8 seconds.	Core average temperature slowly increases with increasing reactor power until reactor trip occurs.	As the pressure decreases, more nucleate boiling occurs in the core causing T_{ave} to increase and fuel temperature to decrease. • The positive MTC and negative Doppler cause power to increase.	DNBR is decreasing due to the decreasing RCS pressure.
21-23	Pzr pressure continues decreasing following the rods dropping into the core.	Core average temperature decreases following the rods dropping into the core.	Nuclear power decreases following the rods dropping into the core.	DNBR reaches its minimum value at 22.9 seconds, above the safety analysis limit value and immediately increases thereafter due to decreasing nuclear power and RCS temperatures.
Results	As was stressed earlier, this analysis stops following the reactor trip. It assumes that the valve will be isolated following the reactor trip and the RCS depressurization will be stopped. The objective of the analysis was to show that DNB would not occur prior to the reactor trip.			

Radiological Consequences

The radiological consequences for the inadvertent opening of a Pzr safety or relief valve are minimal.

Even assuming a direct release to the containment atmosphere, the radiological consequences of such an event are substantially less than that of a LOCA because:

- less primary coolant is released, and
- the activity is lower since no fuel damage is postulated during the event.

Conclusions

The results of the analysis show that the OTΔT reactor trip signal provides adequate protection against the RCS depressurization event.

Exam Bank No.: 2151 **Last used on an NRC exam:** Never

RO Sequence Number: 18

Given the following:

- Unit 2 is operating at 100% power.
- A small Reactor Coolant System leak identified from a Containment Area Rad Monitor develops into a leak exceeding the capacity of a centrifugal charging pump within a two hour period.

Which of the following sets of procedures will be used to directly address this condition?

A. System Operating (POP02) and Emergency (POP05) procedures

B. General Operating (POP03) and Enhanced Off-Normal (POP04) procedures

C. General Operating (POP03) and Emergency (POP05) procedures

D. Enhanced Off-Normal (POP04) and Emergency (POP05) procedures

Answer: D Enhanced Off-Normal (POP04) and Emergency (POP05) procedures

Exam Bank No.: 2151

K/A Catalog Number: EPE 009 G2.2.38 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(10)

Small Break LOCA: Knowledge of conditions and limitations in the facility license.

STP Lesson: LOT 505.02 **Objective Number:** 92115

GIVEN a list of Emergency Operating Procedures, Off Normal Operating Procedures, Annunciator Response Procedures, and Operating Procedures, ARRANGE them in order of hierarchy.

Reference: LOT505.02, lesson plan page 6

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - This distractor is credible because the Emergency Procedures (POP05) do refer to System Operating procedures but would not be directly used to address the RCS leak.

B: INCORRECT - This is a credible distractor because there are two centrifugal charging pumps and it would be reasonable to think that starting the second CCP would provide enough volume to maintain PZR level allowing a normal shutdown of the unit with a General Operating procedure (POP03). But one must remember that a leak greater than the capacity of just one CCP also affects make up capabilty to the VCT and thereby requires entry into the emergency procedures (POP05). Entering the Enhanced Off-Normal procedure (POP04) is correct.

C: INCORRECT - This is a credible distractor because there are two centrifugal charging pumps and it would be reasonable to think that starting the second CCP would provide enough volume to maintain PZR level allowing a normal shutdown of the unit with a General Operating procedure (POP03). But one must remember that a leak greater than the capacity of just one CCP also affects make up capabilty to the VCT. Entering the Emergency procedure (POP05) is correct.

D: CORRECT - Once indication of RCS leakage identified from an area rad monitor, an Enhanced Off-Normal procedure (POP04) will be entered. A leak larger than the capacity of a charging pump will eventually require a manual SI due to inability to maintain PZR level which in turn will require entry into the Emergency procedures (POP05).

Question Level: H **Question Difficulty** 3

Justification:

From the given information, the applicant must determine that a manual SI will be required and then with their knowledge of procedure hierarchy determine how the event will be addressed.

INSTR. NOTE:	Obj. 92115

4.0 PROCEDURE HEIRARCHY

4.1 The highest priority procedure is to be the primary procedure in use. Other procedures may be used in conjunction with the primary procedure provided there is no conflict. The hierarchy is:

4.1.1 Emergency Operating Procedures (EOP) (0POP05s)

4.1.2 Off Normal Operating Procedures (ONP0 (0POP04s)

4.1.3 Annunciator Response Procedures (ARP) (0POP09s)

4.1.4 Normal Operating Procedures (OP) (0POP02s, 0POP03s)

INSTR. NOTE:	If the Control Room has been evacuated, then Control Room Evacuation procedure, 0POP04-ZO-0001, takes precedence over the EOPs and 0POP04-ZO-0008 and 0POP04-ZO-0009. If a fire occurs in Fire Areas 02-78, then 0POP04-ZO-0009, Safe Shutdown Fire Response takes precedence over all EOPs.
INSTR. NOTE:	Obj. 92116

5.0 PROCEDURE USE AND ADHERENCE

5.1 Review “PURPOSE” and “SYMPTOMS OR ENTRY CONDITIONS” to verify ONP is appropriate. (example: Annunciator alarm determined to be not valid)

5.2 May be entered for any of the following reasons:

- A condition is present which is specified in the “SYMPTOMS OR ENTRY CONDITIONS” SECTION.

Exam Bank No.: 2152 **Last used on an NRC exam:** Never

RO Sequence Number: 19

Given the following:

- Unit 2 is operating at 100% power
- 0POP04-RC-0002, Reactor Coolant Pump Off Normal has been entered
- Plant Computer has been lost
- ICS annunciator functions are still working

Under these conditions, the procedure directs the operators to verify the status of RCP oil reservoir annunciators and if necessary, enter the containment and inspect the RCP.

Which of the following is true concerning the performance of these actions?

	Oil Reservoir Annunciator Verification	RCP Inspection
A.	CAN be performed from within the Control Room Horseshoe area	CAN be performed under these plant conditions
B.	CANNOT be performed from within the Control Room Horseshoe area	CAN be performed under these plant conditions
C.	CAN be performed from within the Control Room Horseshoe area	CANNOT be performed under these plant conditions
D.	CANNOT be performed from within the Control Room Horseshoe area	CANNOT be performed under these plant conditions

Answer: C CAN be performed from within the Control Room Horseshoe area; CANNOT be performed under these plant conditions

Exam Bank No.: 2152

K/A Catalog Number: APE 015/017AA1.02 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.8 **10CFR Reference:** 55.41(b)(3)

Ability to operate and / or monitor the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow):
RCP oil reservoir level and alarm indicators

STP Lesson: LOT 201.05 **Objective Number:** 4829

DESCRIBE the instrumentation available for the reactor coolant pumps in the control room and locally

Reference: LOT201.05 handout page 8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - Alarm verification is correct. This is a credible distractor because an RCB entry would be allowed at this power level and is required to inspect RCP oil level but the risk of a radiolagical over exposure is too great at a power level above 10% if entering the boisheild where the RCPs are located, so the Unit must be down powered first.

B: INCORRECT - This distractor is credible because there are numerous alarms that are available only on back panels or plant computer, however RCP reservoir annunciators are located within the horseshoe area. This is a credible distractor because an RCB entry would be allowed at this power level and is required to inspect RCP oil level but the risk of a radiolagical over exposure is too great at a power level above 10% if entering the boishield where the RCPs are located, so the Unit must be down powered first.

C: CORRECT - Annunciators are located on Panel 5 within the control room horseshoe area. RCPs are located within the bioshield in containment which is inaccessible at this power level (procedure has the operators reduce power to less than 10% for RCP inspection).

D: INCORRECT - This distractor is credible because there are numerous alarms that are available only on back panels or plant computer, however RCP reservoir annunciators are located within the horseshoe area. RCB entry portion is correct.

Question Level: F **Question Difficulty** 3

Justification:

Applicant must have knowledge of the annunciator location for RCP oil reservoir levels in the control room and a basic knowledge of containment conditions (radiologically) under these plant conditions.

RCP RELATED REACTOR TRIPS

UnderVoltage sensed on the Class 1E 15 Kv RCP Cubicle (Dummy breaker). Does NOT trip the RCP, but this signal feeds into SSPS and on a 2 of 4 logic, generates a Reactor Trip.

As previously discussed, **Underfrequency** – sensed on the Class 1E 15 Kv RCP Cubicle (Dummy breaker). This signal feeds into SSPS and on a 2 of 4 logic, trips ALL RCPS and also generates a Reactor Trip.

INSTRUMENTATION/ALARMS

The following instrumentation is provided as part of each Reactor Coolant Pump motor:

A. RCP oil pressure switches: (2 per pump)

RCP oil lift pump No. 11 - PS-699A-1/2
RCP oil lift pump No. 12 - PS-699B-1/2
RCP oil lift pump No. 13 - PS-699C-1/2
RCP oil lift pump No. 14 - PS-699D-1/2

The above RCP oil pressure switches are part of an interlock system that prevents starting of a Reactor Coolant Pump until the Lube Oil Lift System is operating properly. The RCP upper thrust bearing must be supplied with lubricating oil at minimum pressure of 600 psig for starting.

When oil pressure gets above 600 psig a blue permissive light (CP-004) comes on.

Administrative controls require that the RCP not be started for at least two minutes after the oil lift pump pressure light has come on. This time delay helps protect against starting the pump with inadequate oil lift pressure, as might be caused by an obstruction in the oil supply line temporarily blocking flow while maintaining pressure above setpoint.

B. Upper Oil Reservoir Liquid Level

A level switch (LS-687A, LS-687B, LS-687C, and LS-687D) is provided in the oil reservoir for the motor upper radial bearing and thrust bearing, with contacts to actuate high and low alarms on CP-005.

The annunciators on CP-005 are as follows:

RCP 1A UPPER OIL RSVR LEVEL HI/LO (1-1) on 1LB005A
RCP 1B UPPER OIL RSVR LEVEL HI/LO (1-2) on 1LB005A
RCP 1C UPPER OIL RSVR LEVEL HI/LO (1-3) on 1LB005A
RCP 1D UPPER OIL RSVR LEVEL HI/LO (1-4) on 1LB005A

Low oil level signals may indicate leaks in the oil piping or oil reservoir. High oil level signals may indicate cooling water leakage permitting oil to mix with water, or possibly a faulty oil-indicating device.

C. RCP Lower Oil Reservoir Liquid Level

Exam Bank No.: 2153 **Last used on an NRC exam:** Never

RO Sequence Number: 20

Pressurizer backup heaters have been energized to recover from a Pressurizer Pressure Control malfunction that resulted in Pressurizer pressure lowering 30 psig.

During the recovery, ___(1)___ heat is being added to raise the fluid temperature in the Pressurizer and ___(2)___ heat is being added to change saturated liquid into a saturated vapor. It takes ___(3)___ energy to change 1 (one) pound of saturated fluid to saturated vapor than it does to raise 1 (one) pound of saturated liquid 1ºF.

	(1)	(2)	(3)
A.	sensible	latent	less
B.	latent	sensible	less
C.	latent	sensible	more
D.	sensible	latent	more

Answer: D sensible, latent, more

Exam Bank No.: 2153

K/A Catalog Number: APE 027 AK1.03 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(14)

Knowledge of the operational implications of the following concepts as they apply to Pressurizer Pressure Control Malfunctions: Latent heat of vaporization/condensation

STP Lesson: LOT 102.54 **Objective Number:** N99793

Define the following terms: Latent Heat of Vaporization

Reference: LOT102.54 handout page 13

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - This distractor is credible because it shows a lack of fundemental knowledge of the concepts of latent heat of vaprization as it applies to the PZR. This distractor does not have all parts of the statement correctly identified.

B: INCORRECT - This distractor is credible because it shows a lack of fundemental knowledge of the concepts of latent heat of vaprization as it applies to the PZR. This distractor does not have all parts of the statement correctly identified.

C: INCORRECT - This distractor is credible because it shows a lack of fundemental knowledge of the concepts of latent heat of vaprization as it applies to the PZR. This distractor does not have all parts of the statement correctly identified.

D: CORRECT - LATENT heat is added for a phase change and SENSIBLE heat is added for a temperature change. It takes MORE heat energy for a phase change than it does to change temperature.

Question Level: F **Question Difficulty** 3

Justification:

Applicant must have an understanding of the fundamental thermodynamic processes which occur inside the Pressurizer.

The latent heat of fusion is the amount of heat that must be added to change the phase of a solid to a liquid at constant temperature and pressure. The latent heat of fusion can also be thought of as the change in specific enthalpy of the substance when changing phase from a solid to a liquid, at a given temperature and pressure. If a liquid is frozen, the latent heat of fusion represents the amount of heat that must be added to change it to a liquid. The latent heat of fusion for ice is 144 Btu/lb_m at a pressure of one atmosphere and a temperature of 32°F.

As heat is added to a liquid at some constant pressure below critical pressure (P_c), we move to the right on the P-T diagram as the temperature of the liquid is elevated, until the Vaporization Line is reached. At this point, any addition of heat results in a phase change as the liquid evaporates into a gas. The latent heat of vaporization is the amount of heat that must be added to cause this phase transition. The change in enthalpy per lb_m of the substance when changing phase from a liquid to a gas, at a constant temperature and pressure, is equal to the latent heat of vaporization. The latent heat of vaporization for water is 970 Btu/lb_m at a pressure of one atmosphere and a temperature of 212°F. Less heat is required at higher temperatures and pressures to produce the same phase change. For example, water at a pressure of 1,000 psia requires the addition of only 650.4 Btu to vaporize one lb_m of water. The saturated steam tables contain the values for latent heat of vaporization for water.

The points at which a substance can exist as both a liquid and gas in equilibrium is represented by the Vaporization Line. Although the Fusion Line has no upper limit, the Vaporization Line terminates at a point defined as the "critical point" for the substance.

The highest temperature (critical temperature) and pressure (critical pressure) at which a gas and liquid can exist in equilibrium as distinguishable phases is represented by the critical point. At temperatures and pressures higher than the critical point, the substance is considered a fluid; something neither gas or liquid. At pressures lower than the critical pressure (but at higher temps), the substance is considered a gas

No definable phase change occurs above the critical point. Two rather arbitrarily drawn lines are extended to the right and upward from the critical point to constitute an area where a gas or liquid state is not readily apparent. Any substance whose property values cause it to fall within this area is referred to as a fluid. A fluid is neither gas nor liquid. A phase transition does not occur at the points that define these lines, but they do correspond to an arbitrary definition of what is a liquid and what is a gas. The fluid region simply resolves the indeterminate area in between these states.

The critical point of water occurs at a pressure of 3,208.2 psia and a temperature of 705.47°F. At the critical point, the latent heat of vaporization is zero, since steam and water are perceived as one and the same. Most atmospheric gases have critical temperatures much lower than water. For example, the critical temperature of helium is 9.54°R or -450.46°F. Therefore, helium at room temperature is at a temperature approximately 55 times greater than its critical temperature. Conversely, the critical temperature of metals is typically much higher than the critical temperature of water.

The single point at which the three phase lines come together is called the "triple point" for the substance. This single point is unique because all three phases (solid, liquid, and gas) can exist in equilibrium with each other while at this pressure and temperature. For example, the triple point of water is at a temperature of 32.02°F and a pressure of 0.089 psia. At this state point, ice, water, and water vapor would exist together.

Exam Bank No.: 2156 **Last used on an NRC exam:** Never

RO Sequence Number: 21

Given the following:

- Unit 1 was operating at 100% power.
- A Main Steam leak occurred inside containment.
- Containment reached a peak pressure of 8 psig.
- Containment pressure and temperature are now slowly lowering.

Which of the following describes the reason for the current containment pressure and temperature response?

Heat is being removed from the containment atmosphere by …

A. both the Containment Spray System and the RCB Chilled Water System flowing through the Reactor Containment Fan Cooler (RCFC) cooling coils.

B. only the RCB Chilled Water System flowing through the Reactor Containment Fan Cooler (RCFC) cooling coils.

C. both the Containment Spray System and the Component Cooling Water System flowing through the Reactor Containment Fan Cooler (RCFC) cooling coils.

D. only the Component Cooling Water System flowing through the Reactor Containment Fan Cooler (RCFC) cooling coils.

Answer: D only the Component Cooling Water System flowing through the Reactor Containment Fan Cooler (RCFC) cooling coils.

Exam Bank No.: 2156

K/A Catalog Number: APE 040 AK3.06 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(7)

Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: Containment temperature and pressure considerations

STP Lesson: LOT 201.12 **Objective Number:** 57126

DESCRIBE the operation of the Component Cooling Water System and its major components. Include automatic actions, interlocks and trips

Reference: LOT201.12 handout page 23

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - This distractor is credible because a knowledge of the Containment Spray actuation setpoint of 9.5 psig is required. This distractor is credible because RCB chilled water is normally aligned to the RCFCs but swaps to the CCW system on an SI. (Containment Pressure 3.0 psig)

B: INCORRECT - This distractor is credible because RCB chilled water is normally aligned to the RCFCs but swaps to the CCW system on an SI. (Containment Pressure 3.0 psig)

C: INCORRECT - This distractor is credible because a knowledge of the Containment Spray actuation setpoint of 9.5 psig is required. CCW through the RCFC coils is correct.

D: CORRECT - A Containment Spray actuation has not occurred, but SI has actuated which swaps RCFC cooling to Component Cooling Water.

Question Level: H **Question Difficulty** 3

Justification:

From the given information, the applicant must determine that a Containment Spray actuation has not occurred (happens at 9.5 psig) and that an SI actuation has occurred (3 psig). The applicant must also know that cooling for the RCFCs swaps from the normal chilled water supply to Component Cooling water upon an SI actuation.

4.7.19 Post-Accident Sampling System

CCW provides cooling water to post-accident sample coolers.

4.7.20 ESF Status Monitoring

Monitors safety-related components of the CCWS and provides alarms and status indication for inoperable equipment, failure to actuate and equipment bypassed conditions.

5.0 SAFETY INJECTION

Upon receipt of the safety injection signal, the following operations are automatically initiated:

5.1 Starting of all three CCW pumps.

5.2 Opening of the CCW heat exchanger outlet valves MOV-0643, MOV-0645, and MOV-0647 and closing of CCW heat exchanger bypass valves MOV-0642, MOV-0644, and MOV-0646. This prevents the CCW heat exchanger from being bypassed thereby ensuring full heat removal capability.

5.3 The isolation valves MOV-0447, MOV-0032, MOV-0235, MOV-0236, MOV-0297, MOV-0392, MOV-0393, FV-4540 and FV-4541 are closed to isolate the following non-ESF components:

- SFP heat exchangers
- BTRS chiller
- LWPS evaporator package
- Letdown heat exchanger
- Excess letdown heat exchanger
- BRS evaporator package
- RCDT heat exchanger
- Primary sample cooler
- Post-accident sample cooler
- Boric acid sample cooler

5.4 The RHR heat exchanger isolation valves FV-4531, FV-4548 and FV-4565 open allowing CCW flow to the RHR heat exchangers.

5.5 Switchover of the RCFCs cooling water source from chilled water to CCW, by automatic closing of chilled water supply and return valves and automatic opening of CCW supply and return valves.

5.6 All the safety injection automatic functions can be performed remotely (manually) from the control room should the automatic system fail.

5.7 Should a Phase B containment isolation signal exist, the CCW to and from the containment for the RCPs will also be by automatic closing of MOV-0318, MOV-0291, MOV-0404, MOV-0542, MOV-0403 and AOV-4493.

5.8 Recirculation Phase

Recirculation mode is the condition after the injection phase. During this mode the CCWS provides cooling to the same components as the injection phase. Two CCW trains are capable of safely cooling down the reactor following a DBA. During the recirculation phase the operator has 2.5 hours to restore the CCW supply to the SFP heat exchangers.

6.0 LOSS OF OFFSITE POWER (LOOP)

The motor-operated valves remain in the same position before the power failure, until the standby diesel generators automatically start to provide power to all CCW safety-related components and valves. As soon as the emergency power is available, all three CCW trains are sequenced on and the chilled water to and from the RCFCs is isolated by closing MOV-0059, MOV-0070, MOV-0137, MOV-0149, MOV-0199, MOV-0209, FV-0864, FV-0852 and FV-0863. CCW to the RCFCs is to be manually provided by the operator within 30 minutes. Additionally MOV-0392 for isolating the RCDT heat exchanger automatically closes. If instrument air is lost during the LOOP, the following pneumatic valves go to their failed position:

- RHR Heat Exchanger Outlet Fails Open.
- Auto Makeup (Demin) Fails Closed.
- Charging pump header cross connects Fail Closed.
- CCW from RCP’s, OCIV FV-4493 Fails Closed.

Exam Bank No.: 2157 **Last used on an NRC exam:** Never

RO Sequence Number: 22

When starting a Reactor Coolant Pump (RCP) in 0POP05-EO-EC11, Loss of Emergency Coolant Recirculation, a Note in the procedure gives the preferred running order of RCP's as follows:

- First - Loop D
- Second - Loop A
- Third - Loops B AND C

Which of the following is true regarding the Note?

The Note helps to .....

A. minimize the effect of RCP operation on RHR Pump performance.

B. minimize the effect of RCP operation on LHSI Pump performance.

C. ensure normal Pressurizer spray flow is available when needed.

D. ensure adequate mixing in all portions of the Reactor Coolant System.

Answer: C ensure normal Pressurizer spray flow is available when needed.

Exam Bank No.: 2157

K/A Catalog Number: EPE E11 G2.4.20 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.8 **10CFR Reference:** 55.41(b)(10)

Loss of Emergency Coolant Recirc: Knowledge of the operational implications of EOP warnings, cautions, and notes.

STP Lesson: LOT 504.27 **Objective Number:** 82520

Given a step, note or caution from 0POP05-EO-EC11, STATE its basis.

Reference: EOPT03.20, page 10

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - Credible because Loop D is the only loop that does not have an RHR pump connected to it and it is listed first, and a running RCP would apply additional back pressure on the discharge of the RHR pump (but not enough to degrade the operation of the RHR pump).

B: INCORRECT - Credible because Loop D is the only loop that does not have an LHSI pump connected to it and it is listed first. However, LHSI pumps injecting and RCPs running would not occur at the same time.

C: CORRECT - The basis for the note is to indicate which combination of RCP(s) will provide normal spray flow.

D: INCORRECT - Credible because each pump produces a different flow distribution within the RCS, however that is not a consideration for the starting order.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have knowledge of procedure basis or an understanding of spray flow dynamics within the RCS.

TITLE LOSS OF EMERGENCY COOLANT RECIRCULATION

EOP NO. 0POP05-EO-EC11 REVISION 18

STP EOP STEP NO.	WOG ERG STEP NO.	BASIS FOR DEVIATION
		The potential for degradation in RCP seal performance and seal life increases with increasing temperature above 300°F. Hence, if RCP seal cooling is lost for a significant period of time, seal and/or bearing damage may occur if temperatures increase above 300°F (16). STP has conservatively determined that the value of 230°F will be used as an operational parameter.
15-Note 1	13-Note	Changed the note to provide a list of what pumps provide normal pressurizer spray (25). This change does not alter the technical intent of the ERG Step and therefore is an allowable ERG (1) deviation.
15-Note 2	N/A	Added note to provide guidance on what constitutes the required number of RCP needed to provide normal spray flow. This enhancement was identified during simulator validation when question arose if both D and A loops were required. Per design calculation NE-PA-92-01-00, only RCP D or RCP A or both RCP B and C can provide spray flow.
15	13	Broke ERG RNO a. into two Substeps under EOP RNO a. using "PERFROM the following:" format per Writers Guide of one action per step. Modified substep c. to direct the user to start a RCP per the normal operating procedure. Steps from the normal operating procedure were not added due to the length of the procedure. Also changed Substep c. to reflect using plant procedure as procedure has pump starting requirements AND has NOTE for preferred order for RCP starting to provide normal pressurizer spray. The above changes do not alter the technical intent of the ERG Step and therefore is an allowable ERG (1) deviation.
16-Note	N/A	A note was added to the procedure to provide notification of a possible ALERT condition. This change is acceptable since it does not alter the technical intent of the ERG (1).

Exam Bank No.: 2165 **Last used on an NRC exam:** Never

RO Sequence Number: 23

Given the following:

- Unit 2 is at 60% power and raising power to 90% at 10%/hour
- Generator Voltage Regulator is ON (auto)
- Inclement weather has caused a grid disturbance
- GEN MAX EXCT alarm on annunciator panel 7M01 illuminates

Which of the following describes the grid disturbance that has occurred and the required operator action in response to the alarm?

A. Grid voltage has risen causing the generator voltage regulator to raise excitation. Lower excitation using the “VOLTAGE ADJUSTER” control.

B. Grid voltage has risen causing the generator voltage regulator to raise excitation. Lower excitation using the “BASE ADJUSTER” control.

C. Grid voltage has lowered causing the generator voltage regulator to raise excitation. Lower excitation using the “VOLTAGE ADJUSTER” control.

D. Grid voltage has lowered causing the generator voltage regulator to raise excitation. Lower excitation using the “BASE ADJUSTER” control.

Answer: C Grid voltage has lowered causing the generator voltage regulator to raise excitation. Lower excitation using the "VOLTAGE ADJUSTER" control.

Exam Bank No.: 2165

K/A Catalog Number: APE 077 AK1.02 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(4)

Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: Over-excitation

STP Lesson: LOT 202.17 **Objective Number:** 91963

DESCRIBE manual and auto voltage regulation.

Reference: 0POP09-AN-07M1, window C-4 (page 19)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - This distractor is credible because rising grid voltage could be thought to also raise excitation on the Main Generator but the opposite is true. Correct control is indicated.

B: INCORRECT - This distractor is credible because rising grid voltage could be thought to also raise excitation on the Main Generator but the opposite is true. This distractor is credible because the Base Adjuster can raise or lower excitation on the Main Generator but not when the Voltage regualtor is ON.and in Auto.

C: CORRECT - Lowering grid voltage will cause the voltage regulator to raise excitation and result in the given alarm. The operator must use the voltage adjuster to lower field current and thereby lower Main Generator excitation.

D: INCORRECT - Lowering grid voltage will cause the voltage regulator to raise excitation and result in the given alarm. This distractor is credible because the Base Adjuster can raise or lower excitation on the Main Generator but not when the Voltage regualtor is ON.and in Auto.

Question Level: H **Question Difficulty** 3

Justification:

From the given conditions, the applicant must determine that Main Generator excitation has risen. Knowledge of generator control is needed to determine that lowering grid voltage will result in the auto voltage regulator raising excitation. The applicant then must know which controller will manually change excitation under these conditions.

GEN MAX EXCT

Automatic Actions: 1) IF the "VOLT REG CONT" switch is in AUTO, THEN exciter current is limited to 112 amps.

Immediate Actions: None

Subsequent Actions: _____ 1) IF the "VOLT REG CONT" switch is in AUTO, THEN PERFORM the following:

_____ a) DECREASE "EXC FLD CUR" until alarm clears, using the "VOLTAGE ADJUSTER" control.

_____ b) ENSURE Main Generator MVARS less than 400 MVAR positive.

CAUTION

Extended manual operation of the Voltage Regulator is not recommended because the exciter limiter circuits are not functional (protection circuits remain functional).

NOTE

- IF "VOLT REG NULL" is NOT at zero, THEN sudden MVAR reading change should be expected when turning "VOLT REG CONT" switch from AUTO to OFF
- IF controlling voltage manually, THEN increased operator awareness is required to maintain MVAR loading on Main Generator.

_____ 2) IF Voltage Regulator output can NOT be reduced in Auto, THEN PERFORM the following:

_____ a) PLACE "VOLT REG CONT" switch in OFF.

_____ b) DECREASE "EXC FLD CUR" until alarm clears, using the "BASE ADJUSTER" control.

_____ c) ENSURE Main Generator MVARS less than 400 MVAR positive.

Exam Bank No.: 2168 **Last used on an NRC exam:** Never

RO Sequence Number: 24

The table below lists Intermediate Range NI readings before and after a unit trip.

	100% power	10 minutes after trip	30 minutes after trip
NI-35	$4x10^{-4}$ amps stable	$8x10^{-10}$ amps slowly lowering	$6x10^{-10}$ amps stable
NI-36	$4.5x10^{-4}$ amps stable	$1.5x10^{-10}$ amps lowering	$2x10^{-11}$ amps stable

Which of the following is true regarding the current status of the Intermediate Range NIs?

A. NI-35 has lost its compensating voltage (under compensated)

B. NI-35 is over compensated

C. NI-36 has lost its compensating voltage (under compensated)

D. NI-36 is over compensated

Answer: A NI-35 has lost its compensating voltage

Exam Bank No.: 2168

K/A Catalog Number: APE 033 AA2.11 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.1 **10CFR Reference:** 55.41(b)(5)

Ability to determine and interpret the following as they apply to the Loss of Intermediate Range Nuclear Instrumentation: Loss of compensating voltage

STP Lesson: LOT 201.16 **Objective Number:** 91250

DEFINE overcompensation and undercompensation and DESCRIBE their effect on intermediate range detector operation.

Reference: LOT201.16, Excore NIs, Powerpoint slide #8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT - An IR detector will normally lower to about 10-10 amps within 10 to 15 minutes following a trip and finally stabilize just above idling current of 10-11 amps. A detector that is reading higher than it should (due to gamma interactions that are not compensated out) is termed to be undercompensated or has lost its compensating voltage.

B: INCORRECT - This distractor is credible because it could be thought that NI-35 is over compensated because it is reading too high.

C: INCORRECT - This distractor is credible because if it were thought that NI-36 was reading too low and if there was a misconception of how compensation works in the Intermediate Range Detectors then it would be under compensated. However, the readings/trends for NI-36 is indicative of normal response following a trip. This detector is correctly compensated.

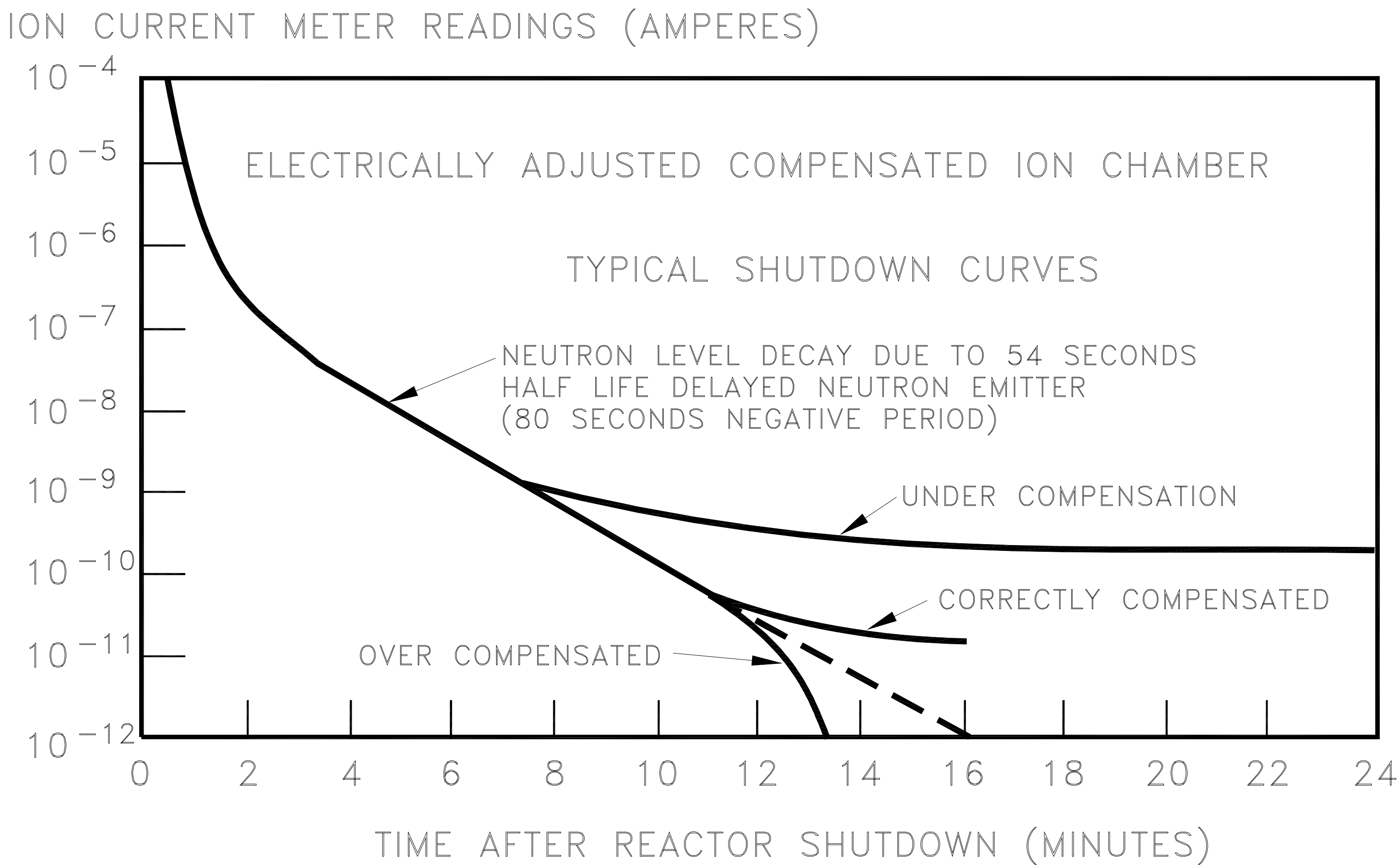
D: INCORRECT - This distractor is credible because if it were thought that NI-36 was reading too low, then the detector would be over compensated. However, the readings/trends for NI-36 is indicative of normal response following a trip. This detector is correctly compensated.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must evaluate the given information and then based on knowledge of normal detector response and definition of terms, determine the correct answer.

GAMMA–COMPENSATED CURVE



LOT201.16.TP.06
DWG-04/05/95

Exam Bank No.: 2170 **Last used on an NRC exam:** Never

RO Sequence Number: 25

0POP05-EO-FRC1 RESPONSE TO INADEQUATE CORE COOLING Step 13 instructs operators to stop ALL Reactor Coolant Pumps prior to depressurizing ALL Intact SG's to Atmospheric Pressure.

Select the RCP parameter that is of concern.

A. RCP motor stator winding temperature

B. RCP lower seal water bearing temperature

C. RCP motor lower radial bearing temperature

D. RCP number 1 seal differential pressure

Answer: D RCP number 1 seal differential pressure

Exam Bank No.: 2170

K/A Catalog Number: EPE 074 EA1.06 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(7)

Ability to operate and monitor the following as they apply to a Inadequate Core Cooling: RCPs

STP Lesson: LOT 504.30 **Objective Number:** 82939

DESCRIBE the indicators available to determine that the RCPs should be stopped.

Reference: WOG FR-C1 Background Document. Pg 42

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: Incorrect - This distractor is credible because the student may associate the containment conditions in this accident with the possible effects on the RCP motor winding temperatures and RCP motor winding temperatures are a for the RCPs..

B: Incorrect - This distractor is credible because the student may associate the RCS conditions in this accident with the possible effects on the RCP lower seal water bearing temperature and RCP lower seal water bearing temperatures are a concern for the RCPs.

C: Incorrect - This distractor is credible because the student may associate the containment conditions in this accident with the possible effects on the RCP motor bearing temperatures and RCP motor bearing temperatures are a concern for the RCPs.

D: Correct - AS the SGs are depressurized the RCS pressure will lower rapidly. The number 1 seal surfaces will come in contact with each other. The RCPs are secured to prevent damage to the seals so that they will be avalible for use later if CETs exceed 1200F

Question Level: F **Question Difficulty** 3

Justification:

The applicant must recognize the effect on the RCS and recall the basis for securing RCPs prior to depressing the SGs.

STEP DESCRIPTION TABLE FOR FR-C.1 Step 13

STEP: Stop All RCPs

PURPOSE: To verify all RCPs have been stopped

BASIS:

In preparation for the subsequent depressurization of the SGs to atmospheric pressure, the RCPs are stopped due to the anticipated loss of Number 1 seal requirements. Continued operation may result in damage to the RCPs.

ACTIONS:

Stop all RCPs

INSTRUMENTATION:

RCP status indication

CONTROL/EQUIPMENT:

RCP switches

KNOWLEDGE:

N/A

PLANT-SPECIFIC INFORMATION:

N/A

Exam Bank No.: 2171 **Last used on an NRC exam:** Never

RO Sequence Number: 26

Given the following:

- Unit 1 is in Mode 6 during a rapid refueling outage
- Core reload has just been completed
- 2 assemblies were placed in the wrong core location during the reload resulting in K_{eff} being higher than predicted

Which of the following is true concerning Shutdown Margin (SDM) as a result of this event?

A. SDM has not changed assuming boron concentration has remained the same.

B. SDM has not changed since all control rods are fully inserted.

C. SDM is larger. The reactor is farther from criticality.

D. SDM is smaller. The reactor is closer to criticality.

Answer: D SDM is smaller. The reactor is closer to criticality.

Exam Bank No.: 2171

K/A Catalog Number: APE 036 AK1.02 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(1)

Knowledge of the operational implications of the following concepts as they apply to Fuel Handling Incidents : SDM

STP Lesson: LOT 201.43 **Objective Number:** 92856

DISCUSS reactivity management concerns when the plant is in Mode 5 & 6 to include:

a. fuel assembly movement

Reference: LOT101.19 handout pages 24 and 29

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: This distractor is credible because the fuel assemblies were put in the wrong place but overall the core still has the same assemblies, however, by placing the assemblies in the incorrect location and it resulted in a higher Keff, the resulting SDM would have to be reduced especially if the Boren concentration did not change even though it would be at about 2800 ppm.

B: INCORRECT: This distractor is credible because the fuel assemblies were put in the wrong place but overall the core still has the same assemblies, however, by placing the assemblies in the incorrect location and it resulted in a higher Keff, the resulting SDM would have to be reduced and in a rapid refueling the control rods would not be inserted but would be locked full out in the Reactor Head.

C: INCORRECT: This distractor is credible because there has been times when students have had misconceptions with how changes in Keff will affect SDM. Placing the assemblies in the incorrect location resulted in Keff higher reducing SDM.

D: CORRECT: If Keff is higher, then SDM would be smaller and the core closer to criticality.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must recall what paramenters affect SDM and how.

REACTIVITY

Thus far the discussion has centered on k_{eff} and the affects on the factors in the six factor formula. Reactor operators, however, do not use k_{eff} very often in referring to the condition of the reactor. There are several reasons that have been given for this. k_{eff} is a multiplication factor, made up by multiplying other parameters together. Reactor operators do not have the luxury of considering only one parameter change at a time. Often, they must consider the affect on the core of many changes that have occurred.

Using the multiplicative nature of k_{eff}, the operator might have to multiply the k_{eff} of having moderator temperature going up, with the k_{eff} associated with rods moving out. This is very awkward. What is needed is something that can be added and subtracted.

Consider a change in rod position and a commensurate change in moderator temperature when operating in the power range (why this happens will be discussed in detail in Chapter 4). The rods move in or out and due to the change in neutron absorption cause a change in the probability of neutrons surviving to cause fission. This creates an imbalance in the core, causing neutron population to change, resulting in a change in moderator temperature. As rods move into the core the neutron absorber in the rod capture more neutrons. A reduction in the neutron population causes reactor power to lower and less heat is generated. This results in a lower RCS moderator temperature.

So, how much rod motion is needed to change temperature by a degree Fahrenheit? A common scale is necessary. Temperature change and rod movement are both changing the conditions in the core, but we have no common scale on which to measure the two effects.

The answer to this problem is a concept called REACTIVITY.

Reactivity is the measure of the departure of a reactor from criticality. Reactivity is defined as the fractional change in neutron population per generation and is indicated by the Greek letter rho (ρ). The fractional change in neutron population per generation (reactivity) can be shown by the equation given below.

$$\rho \equiv \frac{k_{eff} - 1}{k_{eff}}$$

Equation 2-19

The key is that reactivity terms are additive. It is a common scale with which we can quantify the affect on the core due to moderator temperature change as well as the change due to rod motion. It can quantify the change due to boron and xenon concentration changes as well as the effects on the core from power level changes. It is the common scale that the Reactor Operator needs to control reactor power.

Before we can start using it in this way, we need to understand the units used to quantify reactivity.

k_{eff} itself is dimensionless (that is has no units). Thus, reactivity is also dimensionless. Operator use of it in quantifying reactor behavior, however, leads to a need for some sort of dimension. The formula itself is used to define the natural reactivity unit $\Delta k/k$. Also used are the units $\%\Delta k/k$ and pcm as follows:

$$\Delta k_{eff} = k_{eff} - 1$$

$$\rho = \frac{k_{eff} - 1}{k_{eff}} = \frac{\Delta k_{eff}}{k_{eff}} = \frac{\Delta k}{k}$$

Equation 2-20

SHUTDOWN MARGIN

Shutdown margin (SDM) is the instantaneous amount of reactivity that the core is, or can be made, subcritical from its present condition with the most reactive control rod fully withdrawn from the core at any time during the core cycle. Technical Specifications require a shutdown margin with the most reactive rod withdrawn from the core. A typical value required is a shutdown margin of 1.3% Δk/k. These values change depending on the operational mode.

The shutdown margin is calculated using the following equation.

$$\mathrm{SDM} = \frac{1 - k_{eff}}{k_{eff}} = -\rho$$

Equation 2-25

Note that this equation may look like the reactivity equation, but the equation is different; the terms in the numerator are reversed.

Calculate the shutdown margin of a shutdown reactor with a core reactivity value of –0.0045 Δk/k.

Example 2-18

NOTE: Most of the discussion of Shut Down Margin (SDM) is beyond the scope of what has been covered thus far in this course. While a brief discussion of SDM is included here, the main discussion is found in Chapter 8, Reactor Operational Physics.

Exam Bank No.: 2173 **Last used on an NRC exam:** Never

RO Sequence Number: 27

With the plant operating at 100% power, which of the following would violate the definition of containment integrity?

A loss of containment integrity would occur if …..

A. the Supplementary Containment Purge exhaust OCIV and ICIV are opened to reduce containment pressure.

B. BOTH doors of the Personnel Airlock (PAL) OR Auxiliary Airlock (AAL) are opened for material passage.

C. a normally closed air operated containment isolation valve for RCS sampling is opened for chemistry to grab a sample.

D. an automatic containment isolation valve is closed and de-energized for maintenance on the control circuit.

Answer: B BOTH doors of the Personnel Airlock (PAL) OR Auxiliary Airlock (AAL) are opened for material passage.

Exam Bank No.: 2173

K/A Catalog Number: APE 069 AK2.03 **Tier:** 1 **Group/Category:** 2

RO Importance: 2.8 **10CFR Reference:** 55.41(b)(7)

Knowledge of the interrelations between the Loss of Containment Integrity and the following: Personnel access hatch and emergency access hatch

STP Lesson: LOT 503.01 **Objective Number:** 92101

From memory, DEFINE terms used in the Technical Specifications and the Technical Requirements Manual (TRM).

Reference: Tech Spec definition 1.7, Containment Integrity (page 1-2)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - This distractor is credible because normal containment purge valves cannot be opened at power but the supplemental purge system valves automatically close when required.

B: CORRECT - Each airlock must be operable (no more than 1 door open at a time)

C: INCORRECT - This distractor is credible because manual containment isolation valves cannot be opened without affecting containment integrity but air operated isolation valves will automatically close when required.

D: INCORRECT - This distractor is credible because working on containment isolation valves can affect containment integrity but if an MOV is in its required position and de-energized then containment integrity would still be satisfied.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have a working knowledge of the definition of containment integrity.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

a. All penetrations required to be closed during accident conditions are either:

   1) Capable of being closed by an OPERABLE containment automatic isolation valve system, or

   2) Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Specification 3.6.3.

b. All equipment hatches are closed and sealed,

c. Each air lock is in compliance with the requirements of Specification 3.6.1.3,

d. The containment leakage rates are within the limits of Specification 3.6.1.2, and

e. The sealing mechanism associated with each penetration (e.g., welds, bellows, or O-rings) is OPERABLE.

CONTROLLED LEAKAGE

1.8 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

CORE ALTERATIONS

1.9 CORE ALTERATIONS shall be the movement of any fuel, sources, or reactivity control components [excluding rod cluster control assemblies (RCCAs) locked out in the integrated head package] within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT is the unit-specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.6. Plant operation within these core operating limits is addressed within the individual Specifications.

DIGITAL CHANNEL OPERATIONAL TEST

1.10 A DIGITAL CHANNEL OPERATIONAL TEST shall consist of injecting simulated process data where available or exercising the digital computer hardware using data base manipulation to verify OPERABILITY of alarm, interlock, and/or trip functions.

DOSE EQUIVALENT I-131

1.11 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same Committed Effective Dose Equivalent dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The Committed Effective Dose Equivalent dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion and Ingestion," 1988 (Table 2.1, Exposure-to-Dose Conversion Factors for Inhalation).

Exam Bank No.: 2183 **Last used on an NRC exam:** Never

RO Sequence Number: 28

Given the following:

- A Loss of Offsite Power has occurred.
- ALL 4.16KV ESF busses are powered from their associated Emergency Diesel Generator.
- The Unit supervisor directs the operator to "ENSURE Pressurizer Heaters are ON"

Which of the following (1) list the available Pressurizer Heaters and (2) the correct operator action to energize them?

A. (1) Only A, B and C Pressurizer Heater Groups
(2) Reset ESF Load Sequencers, then turn the Control Room Handswitch to 'ON'

B. (1) Only A and B Pressurizer Heater Groups
(2) Reset ESF Load Sequencers, then turn the Control Room Handswitch to 'ON'

C. (1) Only A, B and C Pressurizer Heater Groups
(2) Reset ESF Load Sequencers, then cycle Control Room Handswitch 'OFF' then 'ON'

D. (1) Only A and B Pressurizer Heater Groups
(2) Reset ESF Load Sequencers, then cycle Control Room Handswitch 'OFF' then 'ON'

Answer: D (1) Only A and B Pressurizer Heater Groups
(2) Reset ESF Load Sequencers, then Cycle Contro lroom Handswitch 'OFF' then 'ON'

Exam Bank No.: 2183

K/A Catalog Number: APE 056 AA2.17 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(7)

Ability to determine and interpret the following as they apply to the Loss of Offsite Power: Operational status of PZR backup heaters.

STP Lesson: LOT 201.14 **Objective Number:** 80414

STATE the pressurizer pressure and level control system actuation signals, setpoints, logic, coincidence, and interlocks.

Reference: LOT201.14 PowerPoint slides 9 and 10

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Part 1 is credible since different groups of heaters have different power supplies, so the applicant must have specific knowledge of system design to determine the correct response. Part 2 is credible since most components will change state by going to ON with their control switch. It must be known that an additional interlock exists.

B: INCORRECT: Part 2 is credible since most components will change state by going to ON with their control switch. It must be known that an additional interlock exists.

C: INCORRECT: Part 1 is credible since different groups of heaters have different power supplies, so the applicant must have specific knowledge of system design to determine the correct response.

D: CORRECT: The A and B Pressurizer Heaters would have power available and resetting the ESF Load Sequencer and cycling the control room handswitch would enable Pressurizer Heaters A and B.

Question Level: F **Question Difficulty** 3

Justification:

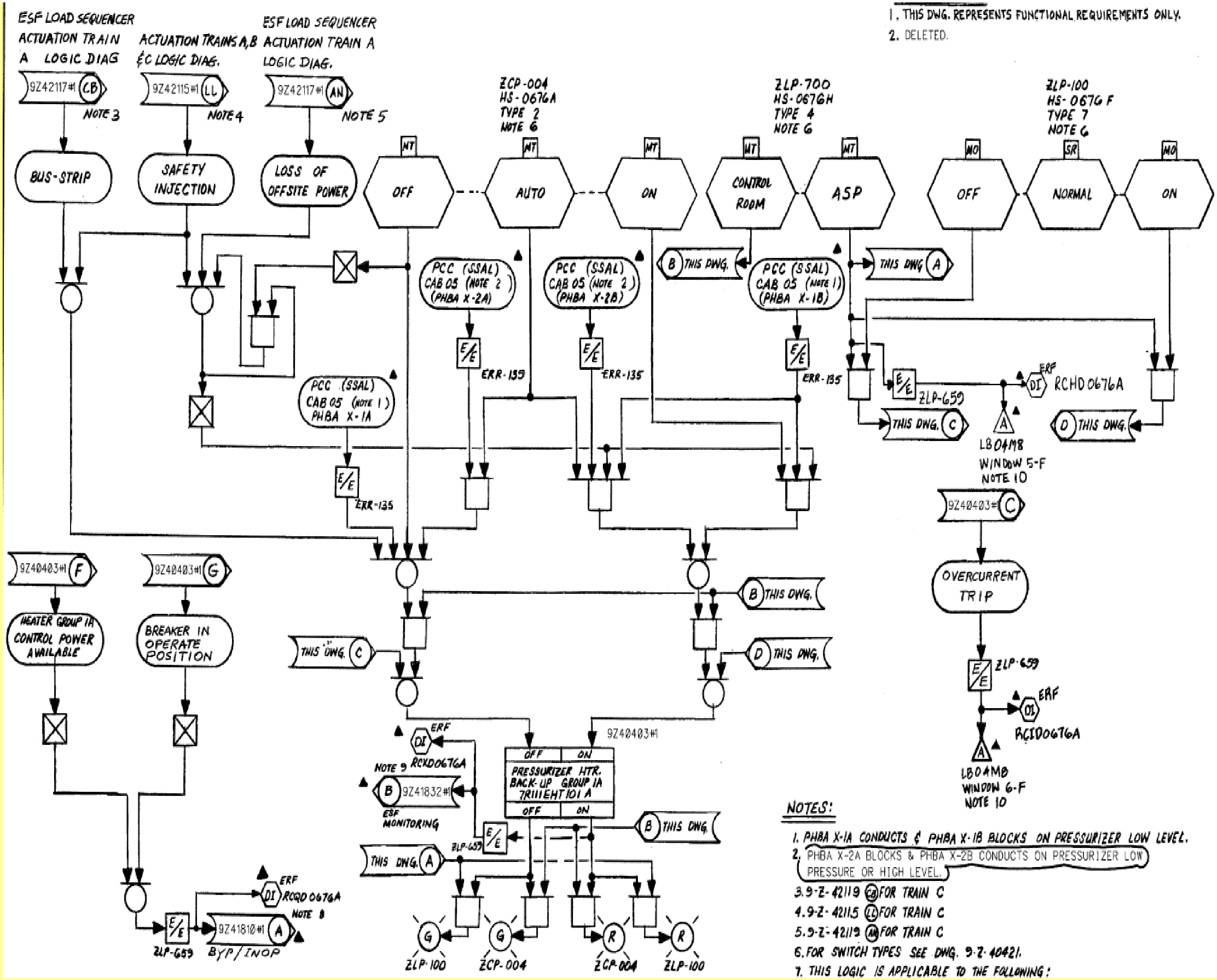
The Reactor Operator must have knowledge of the logics associated with the Pressurizer Heaters.

Pressurizer Pressure and Level Control System

Objective 2: Heater power supplies and KW ratings

PZR HTR B/U GRP A	LC-E1A1	431 KW
PZR HTR B/U GRP B	LC-E1C1	431 KW
CONTROL GROUP C	LC-1N	485 KW
PZR HTR B/U GRP D	LC-1P	377 KW
PZR HTR B/U GRP E	LC-1J2	377 KW
	TOTAL	2101 KW

1. THIS DWG. REPRESENTS FUNCTIONAL REQUIREMENTS ONLY.
2. DELETED.



NOTES:

1. PHBA X-1A CONDUCTS & PHBA X-1B BLOCKS ON PRESSURIZER LOW LEVEL.
2. PHBA X-2A BLOCKS & PHBA X-2B CONDUCTS ON PRESSURIZER LOW PRESSURE OR HIGH LEVEL.
3. 9-Z-42119 CB FOR TRAIN C
4. 9-Z-42115 LL FOR TRAIN C
5. 9-Z-42119 AN FOR TRAIN C
6. FOR SWITCH TYPES SEE DWG. 9-Z-40421.
7. THIS LOGIC IS APPLICABLE TO THE FOLLOWING:

Exam Bank No.: 2210 **Last used on an NRC exam:** Never

RO Sequence Number: 29

With a containment purge in progress in Mode 3, high radiation in containment caused a high alarm on RT-8012, RCB Purge Exhaust.

A note in 0POP04-RA-0001, Radiation Monitoring System Alarm Response, states the following:

A high alarm on RT-8012 or RT-8013 will cause a Containment Ventilation Isolation (CVI). This, in turn, causes RT-8011 sample lines to be isolated and renders RT-8011 radiation monitor inoperable.

Which of the following describes the implications of the note?

With the RT-8011 sample lines isolated, the sample pump …

A. may run indefinitely on recirculation flow. RT-8011 is inoperable; however Technical Specifications are not affected.

B. must be secured to prevent damage. RT-8011 is inoperable; however Technical Specifications are not affected.

C. may run indefinitely on recirculation flow. RT-8011 is inoperable and Technical Specification entry will be required.

D. must be secured to prevent damage. RT-8011 is inoperable and Technical Specification entry will be required.

Answer: D must be secured to prevent damage. RT-8011 is inoperable and Technical Specification entry will be required.

Exam Bank No.: 2210

K/A Catalog Number: EPE 016 G2.4.20 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.8 **10CFR Reference:** 55.41(b)(10)

High Containment Radiation: Knowledge of the operational implications of EOP warnings, cautions, and notes.

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: POP04-RA-0001, page 20 and 105

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Sample pump status is credible because most pumps have a recirculation flowpath, but this sample pump has no recirculation flowpath. Tech Spec is credible because a component either is or is not contained in Tech Specs, specific knowledge is needed to make this determination.

B: INCORRECT: Tech Spec is credible because a component either is or is not contained in Tech Specs, specific knowledge is needed to make this determination.

C: INCORRECT: Sample pump status is credible because most pumps have a recirculation flowpath, but this sample pump has no recirculation flowpath.

D: CORRECT: Must ensure the pump is secured to prevent damage. The radiation monitor is required by the RCS Leakage Detection Tech Spec.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have a knowledge of the operation of the radiation monitor and Tech Spec entry conditions.

NOTE

A high alarm on RT-8012 or RT-8013 will cause a Containment Ventilation Isolation (CVI). This, in turn, causes RT-8011 sample lines to be isolated and renders RT-8011 radiation monitor inoperable.

_____ **3.0 ENSURE RT-8011 RCB Atmosphere Radiation Monitor Sample Pump – STOPPED**

_____ **4.0 REFER TO Technical Specifications (TS) 3.3.2 And 3.4.6.1 For Further Actions**

_____ **5.0 CHECK For Increased Readings On The Following Radiation Monitors:**

- RT-8010A RT-8010B
- RT-8050 RT-8051 RE-8052
- RE-8053 RE-8054 RE-8055
- RE-8056 RE-8099

_____ **6.0 NOTIFY Chemistry And Health Physics To Obtain Grab Samples As Needed To Confirm Increased Readings On Radiation Monitor RT-8012 Or RT-8013**

STEP DESCRIPTION FOR 0POP04-RA-0001 Addendum 2

ADDENDUM: RT-8011 RCB Atmosphere

PURPOSE: To provide the operator with response steps specific to the radiation monitor(s).

BASIS: Due to the different applications provided by the radiation monitors, i.e., area (ARMS), process and liquid effluent (PERMS), monitors required by technical specifications, TRM, ODCM, automatic actuation's etc., response steps will vary. Depending on which monitor is in alarm the operator performs the steps required for that particular monitor.

ACTIONS: The operator performs the appropriate Addendum steps.

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: An RM-11 computer console is located in the Control Room, health physics, and the TSC. The RM-23 is located in the Control Room. RT-8011 sample isolation valves (MOVs-001, 003, 004, and 006) are located on CP002.

KNOWLEDGE: The RCB Atmospheric Monitor is a Non-Class 1E process and effluent monitor. The monitor has three detectors, one for particulate, one for iodine, and one for noble gas. A high alarm on RT-8012 or RT-8013, RCB purge exhaust radiation monitors will cause a Containment Ventilation Isolation (CVI). The CVI causes the sample valves for RT-8011 to shut and render the monitor inoperable. RT-8011 sample pump should automatically shut down after approximately one minute on a loss of sample flow, however the operator should ensure the sample pump is secured to prevent damage to the sample pump. The Unit Supervisor/Shift Manager may elect not to secure containment purge when it is in progress and the alarm condition does not cause the purge permit requirements to be exceeded. If RT-8011 is not operable then Health Physics will take Grab Samples. Chemistry should also be notified because RT-8011 is used for the RCB Purge Permits.

RT-8011 Particulate monitor is only Required in Modes 1 through 4. (Reference 1.a)

Exam Bank No.: 2213 **Last used on an NRC exam:** Never

RO Sequence Number: 30

Given the following:

- A Loss of Coolant Accident (LOCA) has occurred
- Operators are performing the steps of POP05-EO-EO10, Loss of Reactor or Secondary Coolant
- SG A, B, C and D pressures are 800, 810, 790 and 450 psig respectively and slowly lowering
- LOOP A, B, C, and D Tcold are 450, 435, 330 and 440 °F respectively and slowly lowering
- Reactor Vessel Plenum level indicates 20%
- CETs are approximately 375 °F and slowly lowering

Which of the following indicates the status of natural circulation cooling and the expected action for these conditions?

	Natural Circulation	Action
A.	Adequate natural circulation cooling exists	Reduce number of running ECCS pumps
B.	Adequate natural circulation cooling exists	Maintain current steam flow from SGs
C.	Adequate natural circulation cooling does NOT exist	Maintain operation of ECCS pumps
D.	Adequate natural circulation cooling does NOT exist	Open SG PORVs to raise steam flow

Answer: C Adequate natural circulation does NOT exist; Maintain operation of ECCS pumps

Exam Bank No.: 2213

K/A Catalog Number: EPE 011 EA2.09 **Tier:** 1 **Group/Category:** 1

RO Importance: 4.2 **10CFR Reference:** 55.41(b)(5)

Ability to determine or interpret the following as they apply to a Large Break LOCA: Existence of adequate natural circulation

STP Lesson: LOT 504.25 **Objective Number:** 92230

STATE/IDENTIFY the indications available to determine Reactor Coolant System Natural Circulation cooldown rate.

Reference: LOT102.59 student handout page 20

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: NC status is credible because some of the classic indications exist (CETs lowering, loop temperature lowering), however natural circulation is not present with no fluid in the loops (plenum level 20%). Action is credible since E10 does look at stopping LHSI pumps (but based on RCS pressure).

B: INCORRECT: NC status is credible because some of the classic indications exist (CETs lowering, loop temperature lowering), however natural circulation is not present with no fluid in the loops (plenum level 20%). Action is credible because the indicated action would maintain adequate natural circulation (from part one of the distractor)

C: CORRECT: Without fluid in the loops (based on plenum level), natural circulation cannot exist, therefore the correct action is to maintain cooling via ECCS injection.

D: INCORRECT: Action is credible because that is the action that would be taken if NC is not adequate, but only if loops are full.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must evaluate the given conditions and determine the state of natural circulation. Then based on the determination, select the correct action.

NATURAL CIRCULATION

Natural circulation is a basic thermal hydraulic phenomenon that occurs during a loss of the reactor coolant pumps. Heating and cooling of water changes the density of the coolant. As the density decreases, a given volume of water has less mass. The heated water tends to rise, while the cooled water tends to fall. This is similar to the principles of operation of a hot air balloon. To rise, heat is added to the gas volume of the hot air balloon. As the hot air cools, the hot air balloon sinks. Natural circulation is the mechanism by which the coolant is transferred out of the reactor vessel to the steam generators during accident conditions when force circulation is not available which act as a heat sink.

There are two methods used to cause a fluid flow: forced circulation and natural circulation. Forced circulation requires a pump. Natural circulation requires no mechanical work and no moving parts.

The driving force for natural circulation flow is the difference in density between two adjacent masses of fluid.

Conditions Required For Natural Circulation

Natural circulation will only occur if the correct conditions exist. Even after natural circulation begins, removal of any one of the required conditions will stop the natural circulation. The following conditions must exist for natural circulation:

- A density difference. In all practical systems, this density difference is produced by a temperature difference. The warmer fluid is less dense.
- A height difference. The cooler, denser fluid must be at a higher elevation than the warmer, less dense fluid.
- Fluids in physical contact with each other.

There must be two bodies of fluid at different temperatures. This could also be one body of fluid with areas of different temperatures. The difference in temperature is necessary to cause a density difference in the fluid. The density difference is the driving force for natural circulation flow.

The difference in temperature must be maintained for the natural circulation to continue. Addition of heat by a heat source must exist at the high temperature area. Continuous removal of heat by a heat sink must exist at the low temperature area. Otherwise, the temperatures would equalize and no further circulation would occur.

The warm area must be at a lower elevation than the cool area. A warmer fluid is less dense and will tend to rise, and a cooler fluid is denser and will tend to sink. The greater the elevation differences between the warm and the cool fluid masses, the greater the natural circulation flow rate. This is referred to as the heat sink being above the heat source.

Exam Bank No.: 164 **Last used on an NRC exam:** Never

RO Sequence Number: 31

An EOP mitigation strategy to ensure a secondary heat sink is to maintain a minimum AFW flow of 576 gpm.

In accordance with 0POP05-EO-EO00, which of the following cases allows AFW flow to be reduced to Less Than 576 gpm?

	SG NR Level	Ctmt Press.	Ctmt Rad.
A.	A-30% C-24% B-29% D-33%	7 psig	1 R/hr
B.	A-3% C-17% B-2% D-6%	3 psig	10 R/hr
C.	A-17% C-18% B-14% D-21%	5 psig	1 x 10^6 R/hr
D.	A-12% C-0% B-13% D-6%	3 psig	1 x 10^4 R/hr

Answer: B A-3%, C-17%, B-2%, D-6%; 3 psig; 10 R/hr

Exam Bank No.: 164

K/A Catalog Number: G2.4.6 **Tier:** 3 **Group/Category:** 4

RO Importance: 3.7 **10CFR Reference:** 55.41(b)(10)

Knowledge of EOP mitigation strategies

STP Lesson: LOT 504.05 **Objective Number:** 80399

From memory, STATE/IDENTIFY how total AFW flow is verified to be sufficient in the event of a Safety Injection and/or Reactor Trip.

Reference: 0POP05-EO-EO00 pages 3 and 40

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: Incorrect: Credible because the criteria consists a specified level in one or more SGs which is dependant upon wether or not adverse containment conditions exist (based on pressure and radiation levels). Knowledge of the level criteria and entry conditions for adverse containment plus the ability to apply these criteria are needed to correctly respond. In this case, flow cannot be reduced because adverse containment conditions exist and a SG level is not at least 34%.

B: Correct: Not in adverse containment (pressure less than 5 psig and radiation levels less than 1E5 R/HR) with at least 1 SG greater than 14% NR so total AFW flow can be reduced to less than 576 gpm

C: Incorrect: Credible because the criteria consists a specified level in one or more SGs which is dependant upon wether or not adverse containment conditions exist (based on pressure and radiation levels). Knowledge of the level criteria and entry conditions for adverse containment plus the ability to apply these criteria are needed to correctly respond. In this case, flow cannot be reduced because adverse containment conditions exist and a SG level is not at least 34%.

D: Incorrect: Credible because the criteria consists a specified level in one or more SGs which is dependant upon wether or not adverse containment conditions exist (based on pressure and radiation levels). Knowledge of the level criteria and entry conditions for adverse containment plus the ability to apply these criteria are needed to correctly respond. In this case, flow cannot be reduced because no SG level is at least 14% (adverse containment conditions do not exist).

Question Level: F **Question Difficulty** 3

Justification:

Student must know the values and parameters used for adverse containment and review information to select correct condition satisfied.

2) The following are symptoms of a REACTOR TRIP:

- o Any reactor trip annunciator lit
- o Rapid lowering in neutron level
- o All shutdown and control banks fully inserted with rod bottom lights lit

3) The following are symptoms that require a REACTOR TRIP AND SAFETY INJECTION, if one has not occurred:

- o Pressurizer Low Pressure, 2/4 channels LESS THAN OR EQUAL TO 1857 PSIG and NOT BLOCKED
- o Containment High Pressure, 2/3 channels GREATER THAN OR EQUAL TO 3 PSIG
- o Low Compensated SG Pressure, 2/3 channels LESS THAN OR EQUAL TO 735 PSIG on any SG and NOT BLOCKED

4) The following are symptoms of a REACTOR TRIP AND SAFETY INJECTION:

- o Any SI annunciator lit
- o SI pumps running
- o Phase A isolation
- o STBY DGs running

5) The following are entry conditions for 0POP05-EO-EO00, REACTOR TRIP OR SAFETY INJECTION:

- o A Reactor Trip resulting from the Manual actuation of 1/2 Reactor Trip handswitches.
- o A Safety Injection resulting from the Manual actuation of 1/2 Safety Injection handswitches.
- o An automatic Reactor Trip or Safety Injection.

ADVERSE CONTAINMENT CONDITIONS

IF any of the following conditions are met, THEN USE adverse containment values:

- o Containment pressure GREATER THAN OR EQUAL TO 5 PSIG.
- o Containment radiation levels GREATER THAN OR EQUAL TO 10^5 R/HR.
- o Containment integrated radiation dose GREATER THAN OR EQUAL TO 10^6 RADS.

ADDENDUM 5
VERIFICATION OF SI EQUIPMENT OPERATION

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

CAUTION

Equipment should NOT be manually loaded on an ESF Bus until the respective ESF Load Sequencer has completed its automatic sequence OR it has been determined that the respective ESF Load Sequencer has failed to operate.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 3	VERIFY AFW system status:	
	___ a. Motor-driven pump - RUNNING	a. 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).
	___ b. Turbine-driven pump - RUNNING	b. Manually OPEN steam supply valves.
___ 4	VERIFY AFW valve alignment - PROPER EMERGENCY ALIGNMENT	Manually ALIGN valves.
___ 5	VERIFY total AFW Flow - GREATER THAN 576 GPM	PERFORM the following: a. Manually START pumps AND ALIGN valves to feed SGs. b. CONTROL AFW flow to maintain NR level BETWEEN 14% [34%] and 50%.

Exam Bank No.: 278 **Last used on an NRC exam:** 2007

RO Sequence Number: 32

A licensed individual has worked the following daytime schedule:

- Primary Operator - 11/6
- Primary Operator - 11/7
- OFF - 11/8
- Training - 11/9
- Training - 11/10
- Training - 11/11

Which of the following correctly identifies the logbook entries the individual is REQUIRED to review per 0POP01-ZQ-0022, Plant Operations Shift Routines, during shift turnover as Primary Operator on 11/12?

Review of Control Room Logbook entries is required…

A. for only the previous 24 hours.

B. for only the previous 48 hours.

C. for only the previous 72 hours.

D. since the individuals last on-shift duty.

Answer: C for only the previous 72 hours.

Exam Bank No.: 278

K/A Catalog Number: G2.1.3 **Tier:** 3 **Group/Category:** 1

RO Importance: 3.7 **10CFR Reference:** 55.41(b)(10)

Knowledge of shift or short-term relief turnover practices.

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-ZQ-0022, R68, step 3.3.4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because the requiremnet consists of a specified time period. While the indicated time period may seem logical, it is not correct.

B: INCORRECT: Credible because the requiremnet consists of a specified time period. While the indicated time period may seem logical, it is not correct.

C: CORRECT - Procedure requires that on-coming personnel review pertinent information (e.g. special instructions and watchstation logbooks) that have been generated since their last on-shift duty or in the last 72 hours, whichever is less.

D: INCORRECT: Credible because the requiremnet consists of a specified time period. While the indicated time period may seem logical, it is not correct.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must evaluate the given data and make a determination regarding required review of logs.

3.3 Shift Turnover

3.3.1 Off-going Watchstanders SHALL complete applicable portions of the Shift Turnover Checklist Form 4, 5, 6, 7, 8 or 19 for the respective watchstation.

3.3.2 Shift turnover SHALL take place at the normal shift watchstations or their designated locations.

3.3.3 Routine business SHOULD NOT be conducted in the control room during the shift turnover process.

3.3.4 On-coming personnel SHALL review pertinent information (e.g. special instructions and watchstation logbooks) that have been generated since their last on-shift duty or in the last 72 hours, whichever is less.

3.3.5 On-coming and off-going Control Room Watchstanders SHALL walkdown the Control Boards and discuss Shift Turnover Checklist items. This discussion should include, but is NOT limited to:

- Plant Operational Mode
- Status of operating systems and components
- Abnormal equipment alignments
- Inoperable equipment
- Equipment under clearance
- Abnormal annunciator status
- Surveillance or equipment work in progress
- Any events occurring during the shift
- Evolutions in progress

3.3.6 During the control board walkdown, on-coming Control Room Watchstanders SHALL scan the panels to ensure normally lit indications are illuminated (Reference 9.7).

3.3.7 Off-going watchstanders SHALL remain on watch until one of the following conditions is satisfied: (Reference 9.49)

- Their watchstation relief is fully aware of plant conditions.
- IF there is NOT an on-coming relief, THEN watch station plant conditions should be turned over to peer watchstation personnel. (e.g. IF the CP Watch is being secured, THEN ENSURE the TGB Watch is fully aware of the CP watchstation conditions.)

3.3.8 On-coming POs SHALL notify Control Room Personnel that they have assumed the watch as soon as Shift Turnover is complete.

3.3.9 The on-coming Fire Brigade Leader SHALL CONTACT the Off-going Fire Brigade Leader for information that will impact STPEGS as related to fire fighting. (Reference Form 19)

Exam Bank No.: 854 **Last used on an NRC exam:** 2010

RO Sequence Number: 33

Given the following:

- Assume that today is January 15 of the current year.
- A Staff RO, maintaining an "active" license, has performed the functions of an RO during one 12 hour shift since January 1.

Which of the following actions will maintain the RO's license in an "active" status in accordance with 0POP01-ZA-0014, Licensed Operator License Maintenance?

A. One more 12 hour shift performing RO functions during January.
Two more 12 hour shifts performing RO functions during February.

B. Two more 12 hour shifts performing RO functions during February.
Two more 12 hour shifts performing RO functions during March.

C. Two more 12 hour shifts performing RO functions during March.
Four more 12 hour shifts performing RO functions during April.

D. Four more 12 hour shifts performing RO functions during April.
Four more 12 hour shift performing RO functions during May.

Answer: B Two more 12 hour shifts performing RO functions during February, Two more 12 hour shifts performing RO functions during March.

Exam Bank No.: 854

K/A Catalog Number: G2.1.4 **Tier:** 3 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(10)

Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license, 10CFR55, etc.

STP Lesson: LOT 507.01 **Objective Number:** 92184

Given the title of an administrative procedure, IDENTIFY the actions that are performed by the control room operator.

Reference: 0POP01-ZA-0014, Rev 25, Step 4.3.1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because an active license is maintained by performing a specified number of watches within a specified time period. Knowledge of the specific requirements and the ability to apply the requirements are needed to correctly respond.

B: CORRECT: Performing this action will result in five watches during the first quarter (meets the requirement).

C: INCORRECT: Credible because an active license is maintained by performing a specified number of watches within a specified time period. Knowledge of the specific requirements and the ability to apply the requirements are needed to correctly respond.

D: INCORRECT: Credible because an active license is maintained by performing a specified number of watches within a specified time period. Knowledge of the specific requirements and the ability to apply the requirements are needed to correctly respond.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must have a knowledge of the license maintenance requirements and be able to apply those requirements to the given conditions in order to determine the correct response.

4.3 Active License Maintenance

> NOTE
>
> The seven 8-hour or five 12-hour shifts SHALL be within the same calendar quarter AND DO NOT have to be on consecutive shift cycle days. The 8 or 12 hours of a shift must be consecutive hours and according to current on-shift scheduled working hours. The shift hours must be in the position of US or SM for SRO Watchstations and Primary/Secondary RO for RO Watchstation.

4.3.1 The requirements for maintaining an active license for the next calendar quarter are met when:

4.3.1.1 An individual has completed seven 8-hour shifts OR five 12-hour shifts within the same calendar quarter. (10CFR55.53 and NUREG 1262)

4.3.1.2 Individual or Supervisor has verified their License and Respirator Physicals are current and valid.

4.3.1.3 An individual is current in Licensed Operator Requalification Training requirements. Licensed Operator Upgrade Training is not a substitute for this requirement.

4.3.2 Only Qual King or the Training Qual Matrix SHALL be used to determine if individuals with active licenses or current STA qualifications may assume the watch.

4.3.3 Operations Administrative Technician, Training Department, or other authorized individual SHALL update the TRDS database as necessary to ensure the database is maintained current.

4.3.3.1 CERT 821 (RO Watchstation Activation/Maintenance)

4.3.3.2 CERT 822 (SRO Watchstation Activation/Maintenance).

Exam Bank No.: 32 **Last used on an NRC exam:** 1999

RO Sequence Number: 34

Given the following:

- A Reactor Trip occurs due to a Loss of Offsite Power
- The ESF Diesel Generators have all started and restored power to their ESF buses
- The Control Room crew has just completed the Immediate Actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection

Which of the following correctly identifies the status of Containment Cooling?

A. The RCFCs are running and CCW is flowing through the cooling coils.

B. The RCFCs are running and there is NO flow through the cooling coils.

C. The RCFCs are NOT running and CCW is flowing through the cooling coils.

D. The RCFCs are NOT running and there is NO flow through the cooling coils.

Answer: B The RCFCs are running and there is NO flow through the cooling coils.

Exam Bank No.: 32

K/A Catalog Number: 022 A1.04 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.2 **10CFR Reference:** 55.41(b)(7)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Cooling water flow.

STP Lesson: LOT 202.33 **Objective Number:** 4967

State the sources of cooling water to the RCFCs and when each is used.

Reference: Logics 9-Z-42041 Rev 7, 9-Z-42042 Rev 7, 9-Z-41630 Rev 9

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible because this is the normal safety configuration following an ESF actuation.

B: CORRECT - RCFCs are started by the sequencer following a LOOP, and cooling flow to the RCFCs is isolated by the sequencer following a LOOP.

C: INCORRECT - Fan status is credible because a component will either start or not start following a LOOP, and RCB pressure/temperature would not be a factor for a period of time, so it would be reasonable to conclude the fans would not be needed immediately following a LOOP. Flow status is credible because the CCW pumps are started (and needed) following a LOOP. Knowledge of system design is required to choose the correct response.

D: INCORRECT - Fan status is credible because a component will either start or not start following a LOOP, and RCB pressure/temperature would not be a factor for a period of time, so it would be reasonable to conclude the fans would not be needed immediately following a LOOP. Knowledge of system design is required to choose the correct response.

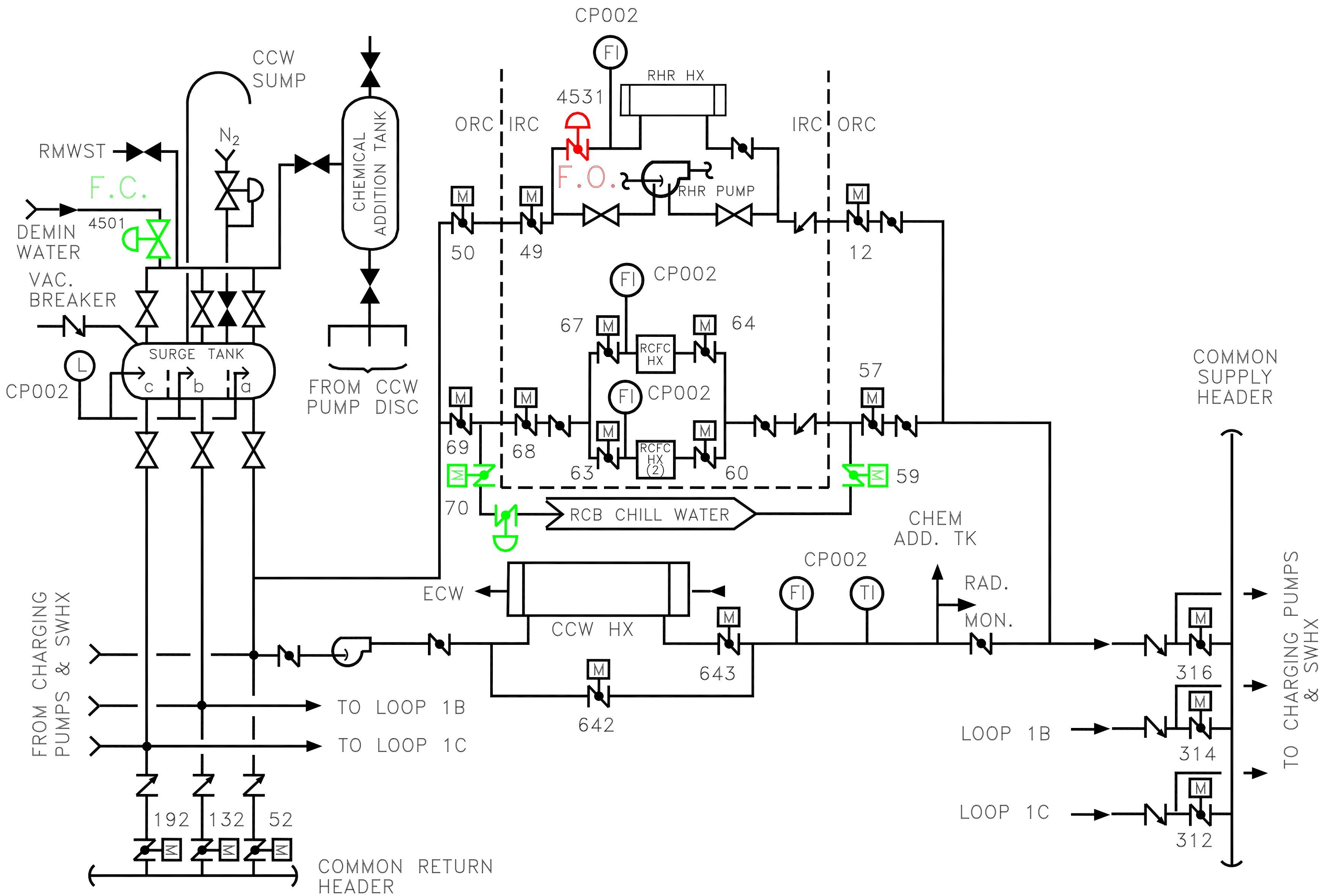
Question Level: H **Question Difficulty** 4

Justification:

The applicant must analyze the given conditions and: 1) Determine that upon a LOOP power is lost to the RCB chill water system which normally supplies cooling to the RCFCs. 2) Determine that the chill water supply valves to the RCFCs are closed and recognize that the CCW supply valves to the RCFCs are closed because an SI signal is not present. Thus, there is no flow through the RCFC cooling coils. 3) Recognize that the RCFCs are sequenced on to their respective ESF buses on a LOOP signal.

CCW ESF LOOP A LOOP

Obj. 7



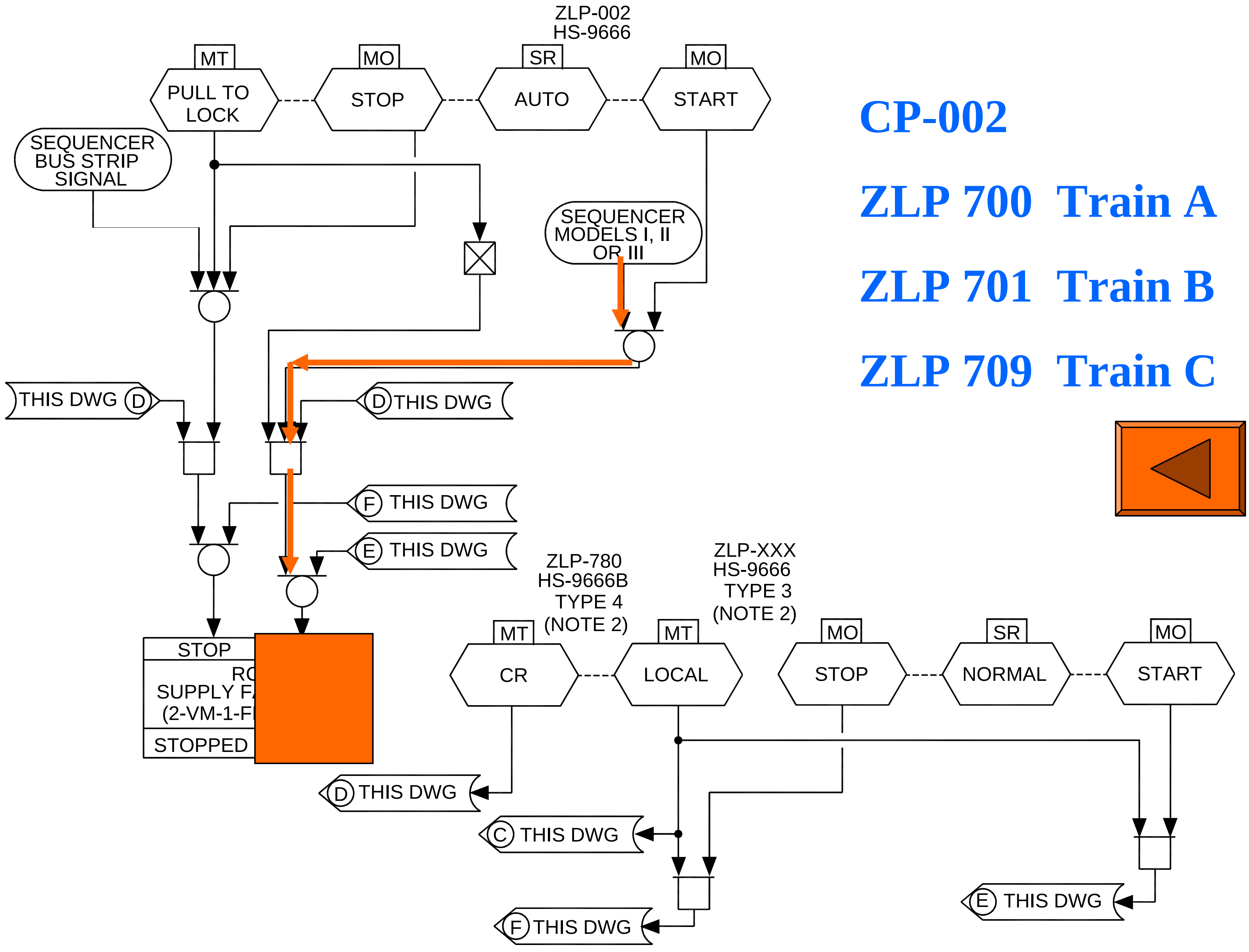
RCFC CONTROL LOGIC

CP-002

ZLP 700 Train A

ZLP 701 Train B

ZLP 709 Train C



LOT202.33.TP.23

Exam Bank No.: 82 **Last used on an NRC exam:** 1999

RO Sequence Number: 35

Unit 1 Train A AND Train C 4.16 KV ESF Busses were de-energized with an expected duration of 7 hours.

Train A 4.16 KV ESF Bus has been re-energized using the Train B ESF Diesel Generator per 0POP04-AE-0004, Loss of Power To One or More 4.16 KV ESF Bus.

Which of the following describes the BASIS for this action?

A. To enable the start of CCP A for RCS Inventory Control.

B. To enable the start of CCW Pump A for Spent Fuel Pool cooling.

C. To extend the use of the Plant Computer System for accident monitoring.

D. To maintain accident monitoring instruments energized by getting a Train A charger in service.

Answer: D To maintain accident monitoring instruments energized by getting a Train A charger in service.

Exam Bank No.: 82

K/A Catalog Number: 062 K1.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(10)

Knowledge of the physical connections and/or cause-effect relationships between the AC distribution system and DC distribution.

STP Lesson: LOT 201.37 **Objective Number:** 92047

State how the Class 1E 125 VDC System interfaces with other systems.

Reference: 0POP04-AE-0004 Rev 13, pages 77 and 88

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible since this would be desireable, however CCP A is powered from E1C (Train C 4.16 KV)

B: INCORRECT - Credible since maintaining SFP cooling is desireable, however any CCW pump can supply SFP cooling and it is given that Train B diesel is available.

C: INCORRECT - Credible since the basis is to maintain accident monitoring instrumentation. However the Plant Computer does not support accident monitors instrumentation nor does it have a Class 1E power supply (QDPS computer system would have been correct).

D: CORRECT - The basis for energizing the Train A bus with the Train B DG is to extend the Train A battery life for accident monitoring instrumentation.

Question Level: F **Question Difficulty** 3

Justification:

Requires a knowledge of the basis for the procedure steps in 0POP04-AE-0004

PROCEDURE PURPOSE

The purpose of this procedure is to restore power to any ESF bus which is not energized. In the case where only one ESF bus is energized by a DG, and another one cannot be energized by the associated DG or offsite power, then steps are taken to operate breakers and disconnects to use the one running DG to supply key loads on another bus.

MAJOR ACTION CATEGORIES

- Tie the operating DG to another bus via the emergency switchgear bus 1L(2L).
- Energize at least one ESF bus from the Emergency Transformer.
- Control and load essential equipment on to the available ESF buses.

DISCUSSION:

STP has committed under specific conditions related to loss of offsite and onsite power to energize at least two ESF buses from a running DG in order to energize specific loads needed to extend station battery life or provide availability of ESF equipment that is electrically powered from one of two specific ESF buses.

STEP DESCRIPTION FOR 0POP04-AE-0004 Addendum 4 AND Addendum 6

STEP: Various

PURPOSE: To provide the actions necessary to energize an ESF bus from an already energized bus via the Emergency Bus 1L(2L).

BASIS: These two Addendum provide the switching instructions necessary to connect either A or C ESF bus to B ESF bus when it is energized from its associated DG. Included are steps for reloading the energized bus and shedding of loads in the event that A or C bus can not be energized. STP is committed in ST-AE-HL-94678 to having a method to energize any ESF bus from an operating DG. At Step 31.0 of Addendum 4, actions are taken to maintain accident monitoring instruments energized by getting a battery charger in service or by shedding non accident monitoring loads. Steps 41.0 and 42.0 of Addendum 4 to isolate air to MSIVs may be needed because of the MSIV Energize to Actuate DCP (Unit 1: DCP 00-01937-90, Unit 2: DCP 00-01937-91). The value of 105.5 VDC to open the battery output breaker comes from Calculation EC-5008, Rev. 13, sheet 252. The Caution uses 105 volts as the minimum and opening at 105.5 provides some margin to equipment damage.

ACTIONS: Energize either A or C ESF bus from B ESF bus via the Emergency Switchgear Bus 1L(2L).

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: Various breakers and disconnects are operated to accomplish this action.

KNOWLEDGE: Due to the planned ESF Bus outages in conjunction with the DG 22 EAOT in 2RE10 outage, steps are included to ensure SFP pump in service for modes 5, 6 and core offloaded to SFP. Having steps to place SFP cooling in service on a loss of power event help the risk profile for 2RE10.

Exam Bank No.: 432 **Last used on an NRC exam:** 1997

RO Sequence Number: 36

A tube leak has occurred in 1D Steam Generator. The Unit is currently performing a rapid plant shutdown for repair.

Under these conditions, failure of which of the following Process and Effluent radiation monitors would allow a release to the environment that would otherwise be automatically prevented?

A. Condenser Air Removal System Monitor (RT-8027)

B. Steam Generator 1D Blowdown Monitor (RT-8025)

C. Turbine Generator Building Drain Monitor (RT-8041)

D. Main Steam Line "D" Monitor (RT-8049)

Answer: C Turbine Generator Building Drain Monitor (RT-8041)

Exam Bank No.: 432

K/A Catalog Number: 073 K3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(7)

Knowledge of effect that a loss or malfunction of the PRM will have on the following: radioactive effluent releases.

STP Lesson: LOT 202.41 **Objective Number:** 80695

STATE the basis for the automatic actions provided by the process radiation monitors.

Reference: 0POP04-RA-0001 page 43

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because this monitor will read upscale during the event, however there is no automatic function associated with this monitor.

B: INCORRECT: Credible because this monitor will read upscale during the event, however there is no automatic function associated with this monitor. There is an automatic function associated with the Steam Generator Blowdown System (RT-8043) that would prevent a release to the environment.

C: CORRECT: Prevents potential release; automatic function stops the TGB Sump #1 pump from discharging to the Oily Waste Storage Tank (outside in the yard).

D: INCORRECT: Credible because this monitor will read upscale during the event, however there is no automatic function associated with this monitor.

Question Level: F **Question Difficulty** 3

Justification:

Applicant must determine from the available distracters which monitor performs an automatic function to prevent a release to the enviroment.

Addendum 15	RT-8041, TGB Drain Monitor	Addendum 15 Page 1 of 1

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1.0	**CHECK HIGH Alarm Exists On Radiation Monitor RT-8041**	GO TO Step 3.0 of this Addendum.
_____ 2.0	**ENSURE TGB Sump Number 1 Sump Pumps 1A(2A), 1B(2B), 1C(2C), And 1D(2D) – STOPPED {TGB 29' west}**	
_____ 3.0	**PERFORM The Following:**	
_____	a. **NOTIFY Chemistry Of The Alarm Condition**	
_____	b. **REQUEST Chemistry Sample The Monitor For Radioactivity**	
_____ 4.0	**NOTIFY Health Physics Of The Alarm Condition**	
_____ 5.0	**GO TO Procedure And Step In Effect**	

Exam Bank No.: 478 **Last used on an NRC exam:** 1995

RO Sequence Number: 37

Conditions have occurred while responding to a Reactor Trip which requires Emergency Boration be initiated. The operator attempts to emergency borate using the Normal Emergency Boration Flowpath, but is unsuccessful at starting a Boric Acid pump.

Which of the following contains two flowpaths, each of which would independently meet the Emergency Boration requirements of 0POP04-CV-0004, Emergency Boration, for the given condition?

A. Boration through Normal Boration Flowpath OR Emergency Boration through "1(2)-CV-0221 MANUAL ALTERNATE IMMEDIATE BORATE"

B. Emergency Boration from RWST OR Emergency Boration via Gravity Feed

C. Emergency Boration through "1(2)-CV-0221 MANUAL ALTERNATE IMMEDIATE BORATE" OR Emergency Boration via Gravity Feed

D. Emergency Boration via Gravity Feed OR Boration through Normal Boration Flowpath

Answer: B Emergency Boration From RWST OR Emergency Boration Via Gravity Feed

Exam Bank No.: 478

K/A Catalog Number: 004 K6.17 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.4 **10CFR Reference:** 55.41(b)(6)

Knowledge of the effect of a loss or malfunction on the following CVCS Components: Flow paths for emergency boration

STP Lesson: LOT 201.07 **Objective Number:** 91060

DESCRIBE the steps necessary to commence an emergency boration in accordance with 0POP04-CV-0003, Emergency Boration.

Reference: 0POP04-CV-0003, Rev 12, Emergency Boration (Pgs 3 & 5)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible because both are valid flowpaths for getting boric acid to the RCS, however Normal Boration Flow Path will not meet the flow requirements without a Boric Acid pump running and Emergency Boration through 1(2) CV-0221Manual Alternate Immediate Borate Valve is not a method listed in the procedure.

B: CORRECT - The methods identified meet the flow requirments and are identified in the procedure.

C: INCORRECT - Credible because Emergency Boration through 1(2) CV-0221Manual Alternate Immediate Borate Valve will get acid to the RCS, but is not a method listed in the procedure. The second method is correct.

D: INCORRECT - First method is correct. The second method is credible because it will get acid to the RCS under normal conditions, but requires a boric acid pump.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must analyze that the inability to start a boric acid pump will mean some of the available flow paths will not provide adequate flow and recall the minimum required flow and alternate flow path identified in the procedure.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 1.0	**CHECK Boric Acid Storage Tank – OPERABLE OR AVAILABLE**	GO TO Addendum 1, Emergency Boration From the RWST.
_____ 2.0	**ENSURE CVCS Charging Line "OCIV MOV-0025" - OPEN**	DISPATCH an operator to open "1(2)-CV-MOV-0025" "CVCS CHG LINE OCIV." {29 ft MAB RM 108C}
_____ 3.0	**ENSURE One Of The Following Valves Are Open:** • Normal Charging "LOOP A ISOL MOV-0003." **OR** • Alternate Charging "LOOP C ISOL MOV-0006."	
____ 4.0	**CHECK Charging Pump - RUNNING**	PERFORM the following: **a.** CLOSE all seal injection OCIVs: • RCP 1A(2A) "SEAL INJ ISOL MOV-0033A" • RCP 1B(2B) "SEAL INJ ISOL MOV-0033B" • RCP 1C(2C) "SEAL INJ ISOL MOV-0033C" • RCP 1D(2D) "SEAL INJ ISOL MOV-0033D" **b.** CLOSE the discharge valve for the Centrifugal Charging Pump to be started: • CCP 1A(2A) "DISCH ISOL MOV-8377A" • CCP 1B(2B) "DISCH ISOL MOV-8377B"

Step 4.0 continued on next page

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 5.0	**PERFORM Both Of The Following:** • START a BA Transfer Pump {CP004} • OPEN "ALT BORATION ISOL MOV-0218" {CP004}	PERFORM all of the following: **a.** COMMENCE boration using the normal boration flowpath. **b.** DISPATCH an operator to perform the following: **1)** IF Boric Acid Tank A is available, THEN DISPATCH an operator to perform the following: • OPEN "1(2)-CV-0333 BORIC ACID TANK 1A(2A) CHARGING PUMP SUCTION ISOLATION VALVE." {19 ft MAB Room 076} • UNLOCK AND OPEN "1(2)-CV-0226 BORIC ACID TANK TO CHARGING PUMP SUCTION BORATION VALVE." {19 ft MAB Room 079}

Step 5.0 continued on next page

Exam Bank No.: 490 **Last used on an NRC exam:** 1995

RO Sequence Number: 38

While operating in Mode 4, annunciator window 5M02-B-7, RCS COLD OVERPRESS ALERT-TRN B, illuminates. The operator notes that COMS has NOT actuated.

Which of the following instrument failures could be the cause of the annunciator?

A. Loop C WIDE range cold leg temperature failed High (TE-434)

B. Loop C WIDE range cold leg temperature failed Low (TE-434)

C. Loop C NARROW range cold leg temperature failed High (TE-430)

D. Loop C NARROW range cold leg temperature failed Low (TE-430)

Answer: B Loop 'C' WR cold leg temperature failed low (TE-434)

Exam Bank No.: 490

K/A Catalog Number: 002 K4.10 **Tier:** 2 **Group/Category:** 2

RO Importance: 4.2 **10CFR Reference:** 55.41(b)(7)

Knowledge of RCS design feature(s) and/or interlock(s) which provide for the following: Overpressure protection

STP Lesson: LOT 201.14 **Objective Number:** 80414

STATE the pressurizer pressure and level control system actuation signals, setpoints, logic, coincidence, and interlocks.

Reference: 0POP04-RP-0005, Rev 13, (Pg 6); LOT 201.14, Rev 14, (Pg 6)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible because temperature is an input. The applicant must understand COMS design/operation to correctly respond.

B: CORRECT - A lowering temperature will lower the COMS lift setpoint and cause the alarm to illuminate (alarm comes in when pressure is within 20 psi of lift setpoint).

C: INCORRECT - Credible because temperature is an input. The applicant must understand COMS design/operation to correctly respond.

D: INCORRECT - Credible because temperature is an input. The applicant must understand COMS design/operation to correctly respond.

Question Level: H **Question Difficulty** 3

Justification:

The applicant has to recall that COMS Train B receives input from auctioneered Lo WR Tcold and that only the Alert alarm is actuated if COMS is not ARMED.

_____ **4.0 CHECK COMS Actuation Due To Instrument Malfunction**

_____ **a. VERIFY following channels - OPERABLE:**

- PT-403, Loop B Wide range pressure (QDPS)
- TE-413, Loop A W/R Hot Leg (QDPS)
  - TR-413 (CP 018)
- TE-423, Loop B W/R Hot Leg (QDPS)
  - TR-423 (CP 018)
- TE-433, Loop C W/R Hot Leg (QDPS)
  - TR-433 (CP 018)
- TE-443, Loop D W/R Hot Leg (QDPS)
  - TR-443 (CP 018)

a. PERFORM the following:

1) PLACE the Cold "OVERPRESSURE MIT" Switch for PCV-0655A in BLOCK.

2) IF PCV-0655A is closed, THEN ENSURE PCV-0655A isolation valve open.

3) ENSURE PCV-0655A handswitch in AUTO.

_____ **b. VERIFY following channels - OPERABLE:**

- PT-404, Loop C Wide range pressure (QDPS)
- TE-414, Loop A W/R Cold Leg (QDPS)
  - TR-413 (CP 018)
- TE-424, Loop B W/R Cold Leg (QDPS)
  - TR-423 (CP 018)
- TE-434, Loop C W/R Cold Leg (QDPS)
  - TR-433 (CP 018)
- TE-444, Loop D W/R Cold Leg (QDPS)
  - TR-433 (CP 018)

b. PERFORM the following:

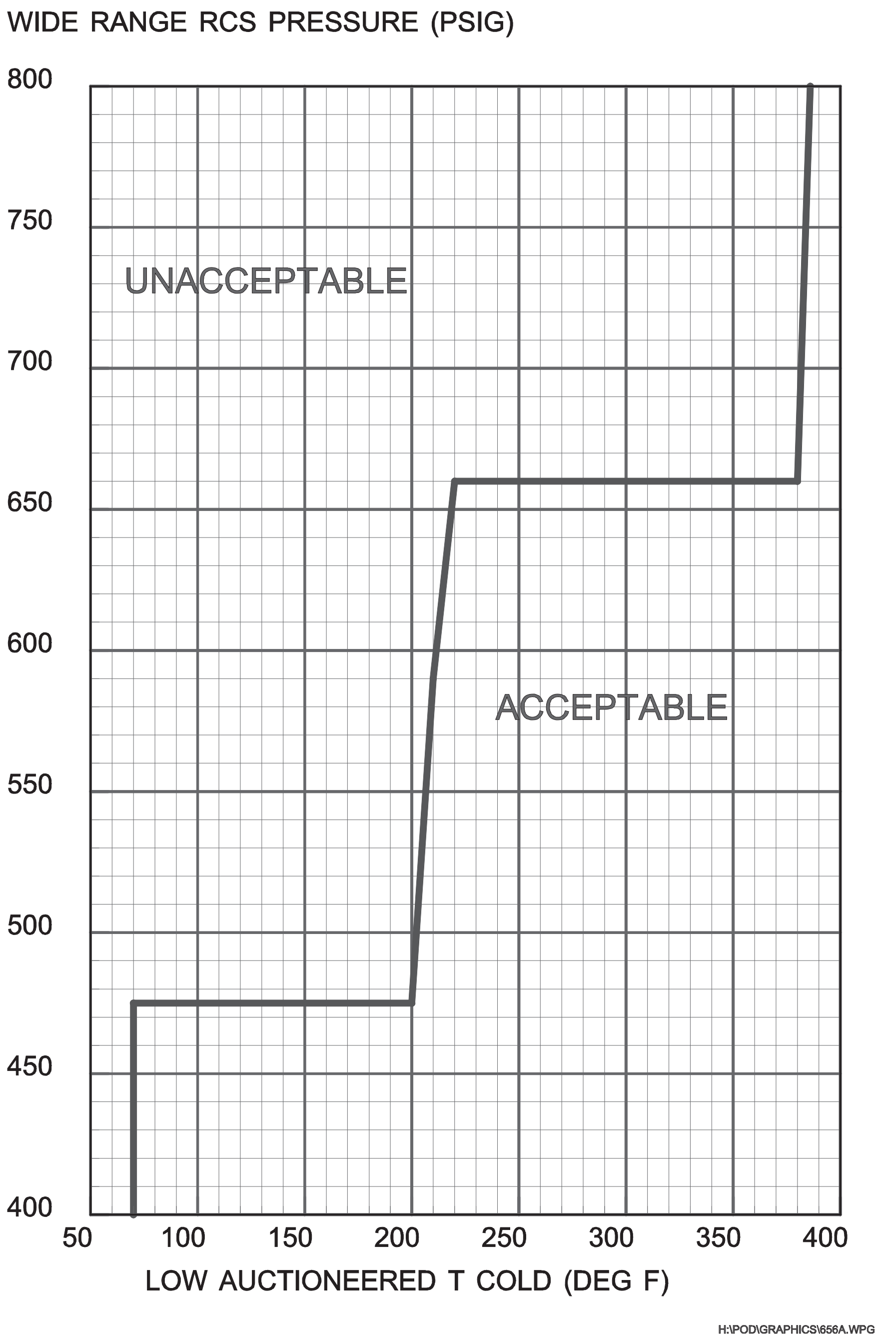
1) PLACE the Cold "OVERPRESSURE MIT" Switch for PCV-0656A in BLOCK.

2) IF PCV-0656A is closed, THEN ENSURE PCV-0656A isolation valve open.

3) ENSURE PCV-0656A handswitch in AUTO.

Addendum 1	Cold Overpressure Limits	Addendum 1 Page 1 of 2

LOW PORV, PCV-0656A



Exam Bank No.: 492 **Last used on an NRC exam:** 1995

RO Sequence Number: 39

What is the minimum configuration of ECCS equipment assumed by the FSAR to inject into the reactor vessel to assure adequate core cooling in the event of the design basis LOCA?

A. Two HHSI pumps, two LHSI pumps, two Accumulators

B. Two HHSI pumps, two LHSI pumps, one Accumulator

C. One HHSI pump, one LHSI pump, two Accumulators

D. One HHSI pump, one LHSI pump, one Accumulator

Answer: C One HHSI pump, one LHSI pump, two Accumulators

Exam Bank No.: 492

K/A Catalog Number: 006 K1.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.2 **10CFR Reference:** 55.41(b)(8)

Knowledge of the physical connections and/or cause-effect relationships between the ECCS and the following systems: RCS

STP Lesson: LOT 201.10 **Objective Number:** 4123

State the function of the ECCS and each of it major components.

Reference: LOT201.10,HO.01 handout page 3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT- Credible because some combination of HHSI, LHSI and acumulators is required. The applicant must have knowledge of these requirements to choose the correct response.

B: INCORRECT- Credible because some combination of HHSI, LHSI and acumulators is required. The applicant must have knowledge of these requirements to choose the correct response.

C: CORRECT - Minimum ECCS flow assumes one train of ECCS fails to start, one train dumps to containment through initiating break, and one train reaches core. Accumulators assume one accumulator dumps to containment through initiating break, and two accumulators reach core.

D: INCORRECT- Credible because some combination of HHSI, LHSI and acumulators is required. The applicant must have knowledge of these requirements to choose the correct response.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must recall the minimum ECCS equipment for the design basis LOCA to ensure core cooling.

In the event of a break which maintains RCS pressure > LHSI pump shut-off head, flow provided from one HHSI pump and two Accumulators is sufficient to meet minimum ESF performance criteria.

In Summary, A minimum of 2 Accumulators delivering to two unaffected loops and one HHSI and one LHSI pump delivering to an unaffected loop will assure adequate core cooling for the design basis LOCA.

Safe Shutdown Assessment

Contained in "STP Fire Hazard Analysis Report" (FHAR) and used to show compliance with the requirements of 10CFR50 Appendix R, section III.G Fire Protection of Safe Shutdown Capability, and III.L, Alternative and dedicated Shutdown Capability.

Safe shutdown analysis assumes pressure control capability during a cooldown/depressurization. During depressurization of the RCS, the SI Accumulators need to be isolated or depressurized to prevent injection and allow the RCS to be depressurized. The Accumulator isolation valves and Accumulator nitrogen venting valve are required for this capability.

If closing their outlet valves cannot isolate the Accumulators; the gas pressure will be vented to the RCB atmosphere. Either way, injection of the contents would be prevented.

FLOWPATHS

The injection phase is defined as period in which borated water is delivered from the RWST and Accumulators to the RCS cold legs.

The ECCS minimizes and prevents core damage by rapidly refilling the reactor vessel and reflooding the core providing short term core cooling and terminates reactivity increases.

During the injection phase, the RWST provides borated water to 3 CS/SIS suctions connecting off the main header.

The HHSI pumps inject when RCS pressure is less than 1600 psig and are normally lined up for cold leg injection. The pumps are provided with miniflow return lines to the RWST to protect running against shut-off head.

The Accumulators inject stored borated water when RCS pressure is 590-670 psig. Accumulator injection pressure is established by nitrogen and they can only inject into the cold legs.

LHSI pumps inject when RCS pressure is less than 300 psig and are normally lined up through the RHR heat exchanger for cold leg injection. The pumps are provided with miniflow returns lines to RWST to protect running against shutoff head.

The cold leg recirculation phase is defined as that period in which borated water is recirculated from containment sumps to the RCS cold legs via LHSI/HHSI pumps.

The cold leg recirculation phase terminates core boiling and is initiated automatically by the Auto-Recirc signal:

Exam Bank No.: 2220 **Last used on an NRC exam:** Never

RO Sequence Number: 40

Emergency Diesel Generator Trip Solenoids have Class 1E and Non-Class control power.

Which of the following states the source of the Trip Solenoids control power?

	Class 1E Control Power	**Non-Class Control Power**
A.	Class 1E 120 Volt Vital AC	Non-Class 125 Volt DC
B.	Class 1E 125 Volt DC	Non-Class 125 Volt DC
C.	Class 1E 120 Volt Vital AC	Non-Class 120 Volt Vital AC
D.	Class 1E 125 Volt DC	Non-Class 120 Volt Vital AC

Answer: B Class 1E 125 Volt DC - Non-Class 125 Volt DC

Exam Bank No.: 2220

K/A Catalog Number: 064 K2.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.2 **10CFR Reference:** 55.41(b)(7)

Emergency Diesel Generators (ED/G)
Knowledge of bus power supplies to the following:
Control Power

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 slide 191

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

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.

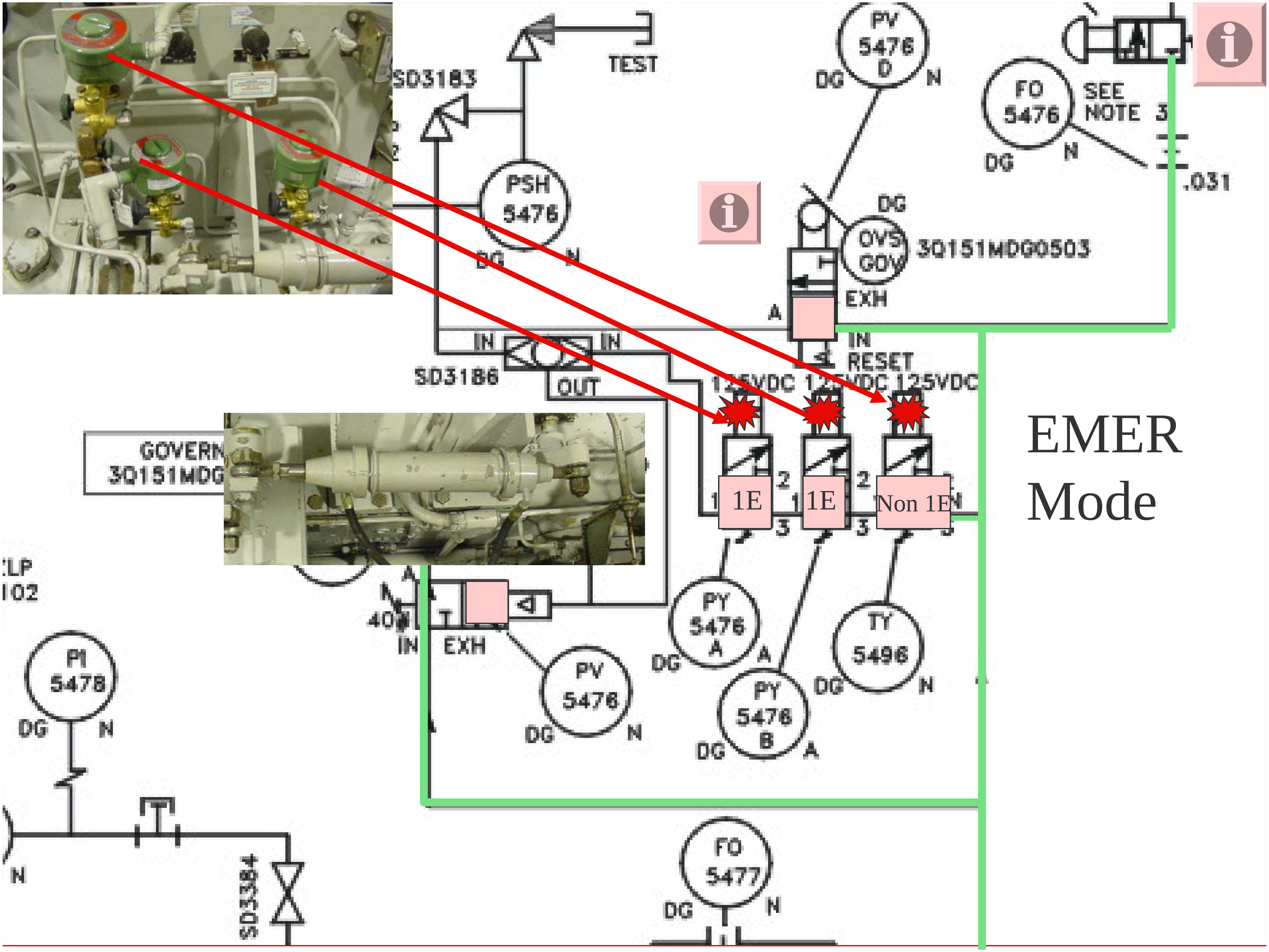
C: INCORRECT: Credible because solenoids can be either AC or DC powered. The applicant must have knowledge of system design to correctly respond.

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.


EMER
Mode
1E
1E
Non 1E
125VDC
125VDC
125VDC
SD3183
TEST
PSH
5476
PV
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D
FO
5476
SEE
NOTE 3
.031
OVS
GOV
3Q151MDG0503
EXH
IN
RESET
SD3186
IN
OUT
GOVERN
3Q151MDG
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5476
A
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5476
B
TY
5496
PV
5476
PI
5478
FO
5477
SD3384
EXH
DG
N

Exam Bank No.: 922 **Last used on an NRC exam:** Never

RO Sequence Number: 41

Upon receipt of a Safety Injection signal, Pressurizer heaters that are supplied by ESF busses are de-energized.

Which of the following describes the Pressurizer heaters that will be de-energized?

A. Proportional Heater Group C and Backup Heaters Group A and B.

B. Proportional Heater Group C and Backup Heaters Group D and E.

C. Only Backup Heaters Group A and B.

D. Only Backup Heaters Group D and E.

Answer: C Only Backup Heaters Group A and B.

Exam Bank No.: 922

K/A Catalog Number: 010 K2.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.0 **10CFR Reference:** 55.41(b)(7)

Knowledge of bus power supplies to the following:
PZR heaters.

STP Lesson: LOT 201.14 **Objective Number:** 8860

List power supplies for pressurizer heaters.

Reference: LOT201.14 handout #1, page 4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - This distractor is credible because it shows a lack of knowledge with PZR Heater power supplies. Only the ESF electrical powered heaters are shed. The control heaters and backup heaters D and E are nonclass power and are only controlled from pressurizer pressure control system and low pressurizer level.

B: INCORRECT - This distractor is credible because it shows a lack of knowledge with PZR Heater power supplies. Only the ESF electrical powered heaters are shed. The control heaters and backup heaters D and E are nonclass power and are only controlled from pressurizer pressure control system and low pressurizer level.

C: CORRECT - The A and B backup heaters are powered from class 1E powered 480V LC E1A1 and E1C1 respectively.

D: INCORRECT - This distractor is credible because it shows a lack of knowledge with PZR Heater power supplies. Only the ESF electrical powered heaters are shed. The control heaters and backup heaters D and E are nonclass power and are only controlled from pressurizer pressure control system and low pressurizer level.

Question Level: F **Question Difficulty** 3

Justification:

The applicant is required to recall that the ESF sequencer strips only backup heaters A and B that powered from ESF electrical busses.

NOTE:	Slide 9

HEATER POWER SUPPLIES AND KW

PZR HTR B/U GRP A	LC-E1A1	431 KW
PZR HTR B/U GRP B	LC-E1C1	431 KW
CONTROL GROUP C	LC-1N	485 KW
PZR HTR B/U GRP D	LC-1P	377 KW
PZR HTR B/U GRP E	LC-1J2	377 KW
	TOTAL	2101 KW

NOTE:	Slide 10

The ESF powered Group A and B heaters will be de-energized on an ESF actuation. After the actuation has been reset, the heater hand switch on CP-004 must be taken to “OFF” to clear the seal-in signal which turned the theaters off. Then the heaters can be energized (providing level in the pressurizer is >17%).

This feature only applies to the ESF powered heaters. This is to prevent the heaters from cycling on when the ESF bus may be powered from the ESF diesel.

NOTE:	Slides 11-14

COLD OVERPRESSURE MITIGATION SYSTEM (COMS)

Provides protection against RCS over-pressurization when temperature is below RT_{NDT}. The system uses the power operated relief valves (PORVs). Inputs to the COMS are:

1. Train A COMS (PCV-655A) which receives signals from auctioneered low RCS wide range T_H and wide range RCS pressure channel 403.

2. Train B COMS (PCV-656A) which receives signals from auctioneered low RCS wide range T_C and wide range RCS pressure channel 404.

Exam Bank No.: 940 **Last used on an NRC exam:** Never

RO Sequence Number: 42

The Reactor tripped and Auxiliary Feedwater has actuated.

Which of the following completes the statement concerning the heat transfer relationship between the RCS and SGs?

The heat transfer rate between the RCS and the SGs will…

A. lower as RCS temperature rises and AFW flow rises.

B. lower as AFW temperature lowers and AFW flow rises.

C. rise as AFW temperature rises and RCS flow lowers.

D. rise as RCS temperature rises and AFW flow rises.

Answer: D rise as RCS temperature rises and AFW flow rises.

Exam Bank No.: 940

K/A Catalog Number: 061 K1.04 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.9 **10CFR Reference:** 55.41(b)(14)

Knowledge of the physical connections and/or cause-effect relationships between the AFW and the following systems: RCS

STP Lesson: LOT 102.58 **Objective Number:** N99867

Describe three mechanisms of heat transfer.

Reference: LOT102.58 GP instructor guide page 8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible because temperature and/or flow changes in either the hot or cold fluids will affect heat transfer rate. The applicant must closely analyze the given conditions to determine if the indicated changes in the two systems will cause the transfer rate to change as indicated.

B: INCORRECT - Credible because temperature and/or flow changes in either the hot or cold fluids will affect heat transfer rate. The applicant must closely analyze the given conditions to determine if the indicated changes in the two systems will cause the transfer rate to change as indicated.

C: INCORRECT - Credible because temperature and/or flow changes in either the hot or cold fluids will affect heat transfer rate. The applicant must closely analyze the given conditions to determine if the indicated changes in the two systems will cause the transfer rate to change as indicated.

D: CORRECT - as the RCS heats up and AFW flow rises, a large delta T is created raising the heat transfer rate.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must recall the cause and effect relationship of differential temprature and heat transfer rate and apply the concept to the given situation.

INSTRUCTOR GUIDE	KEY POINTS, AIDS, QUESTIONS/ANSWERS

Q = total heat added (Btu)

C_p = heat capacity (Btu/°F)

ΔT = change in temperature (°F)

m = mass (lb_m)

c_p = specific heat (Btu/lb_m °F)

B. The British thermal unit (Btu) is unit of heat energy and is defined in terms of these relationships

1. One Btu is defined as amount of heat required to raise temperature of one pound-mass of water at standard atmospheric pressure by one degree Fahrenheit

C. In most power plant applications, heat is added to flowing fluids rather than stagnant bodies

1. For these applications, it is convenient to rewrite these relationships in terms of heat addition or removal rate ($\dot{Q}$) and mass flow rate ($\dot{m}$)

Equation 7-2

$$\dot{Q} = \dot{m}c_p\Delta T$$

Where:

$\dot{Q}$ = heat addition or removal rate (Btu/hr)

$\dot{m}$ = mass flow rate (lb_m/hr)

c_p = specific heat (Btu/lb_m °F)

ΔT = change in temperature (°F)

Example 7-1

Calculate rate of heat addition for a heat exchanger operating with these conditions:

Coolant temperature in = 535°F

Coolant temperature out = 551°F

Coolant flow rate = 7×10^7 lb_m/hr

Exam Bank No.: 1330 **Last used on an NRC exam:** 2011

RO Sequence Number: 43

Which one of the following correctly describes the SEQUENCE of events as Instrument Air pressure lowers from the normal operating value?

A. -Air Compressor 14 (24) starts/loads.
-Service Air Isolation Valve PV-9785 closes.
-Instrument Air to Yard Valve PV-8568 closes.
-Instrument Air Dryer Bypass Valve PV-9983 opens.

B. -Instrument Air to Yard Valve PV-8568 closes.
-Air Compressor 14 (24) starts/loads.
-Instrument Air Dryer Bypass Valve PV-9983 opens.
-Service Air Isolation Valve PV-9785 closes.

C. -Air Compressor 14 (24) starts/loads.
-Instrument Air Dryer Bypass Valve PV-9983 opens.
-Service Air Isolation Valve PV-9785 closes.
-Instrument Air to Yard Valve PV-8568 closes.

D. -Service Air Isolation Valve PV-9785 closes.
-Air Compressor 14 (24) starts/loads.
-Instrument Air Dryer Bypass Valve PV-9983 opens.
-Instrument Air to Yard Valve, PV-8568 closes.

Answer: A -Air Compressor 14 (24) starts/loads.
-Service Air Isolation Valve PV-9785 closes.
-Instrument Air to Yard Valve PV-8568 closes.
-Instrument Air Dryer Bypass Valve PV-9983 opens.

Exam Bank No.: 1330

K/A Catalog Number: 078 K1.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.7 **10CFR Reference:** 55.41(b)(4)

Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: Service Air

STP Lesson: LOT 202.26 **Objective Number:** 92995

Given a scenario in which Instrument Air pressure is decreasing, PREDICT Instrument and Service Air system component automatic actions that will occur as pressure decreases.

Reference: 0POP04-IA-0001, Loss Of Instrument Air, Rev. 16 pg 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT: This is the correct sequence IAW the system off-normal procedure.

B: INCORRECT: Credible because all the actions occur. The applicant must have knowledge of system design/operation to respond correctly.

C: INCORRECT: Credible because all the actions occur. The applicant must have knowledge of system design/operation to respond correctly.

D: INCORRECT: Credible because all the actions occur. The applicant must have knowledge of system design/operation to respond correctly.

Question Level: F **Question Difficulty** 3

Justification:

Applicant must know the automatic actions and setpoints for the IA and SA systems.

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.: 1401 **Last used on an NRC exam:** 2009

RO Sequence Number: 44

Given the following conditions:

- Unit 1 is in Mode 5
- RHR Train ‘A’ is in service providing shutdown cooling.
- FV-8565, IA OCIV, subsequently fails closed.

Which of the following correctly describes the effect of the valve failure?

RHR Train ‘A’ is.....

A. AVAILABLE to provide shutdown cooling since instrument air accumulator tanks in containment will continue to supply the necessary air and allow normal operation.

B. AVAILABLE to provide shutdown cooling since the RHR Heat Exchanger Outlet valve fails open and the RHR Heat Exchanger Bypass valve fails closed providing full cooling flow.

C. NOT available to provide shutdown cooling since the RHR Heat Exchanger Outlet valve fails closed and the RHR Heat Exchanger Bypass valve fails open providing no cooling flow.

D. NOT available to provide shutdown cooling since the RHR Pump Recirculation valve fails open which would not allow adequate cooling water flow to reach the RCS.

Answer: B AVAILABLE to provide shutdown cooling since the RHR Heat Exchanger Outlet valve fails open and the RHR Heat Exchanger Bypass valve fails closed providing full cooling flow.

Exam Bank No.: 1401

K/A Catalog Number: G2.2.37 **Tier:** 3 **Group/Category:** 2

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(7)

Ability to determine operability and/or availability of safety related equipment.

STP Lesson: LOT 201.09 **Objective Number:** 4245

GIVEN a plant or system condition, PREDICT the operation of the Residual Heat Removal system

Reference: LOT201.09, RHR System, PowerPoint slides 14 and 15

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible since some plant air operated valves have air accumulators for increased reliability during loss of air scenarios.

B: CORRECT - The valves fail as indicated, thus providing full cooling flow to the RCS.

C: INCORRECT - Credible since air operated valves can fail in either direction, depending on design. The applicant must understand design/safety considerations for the system.

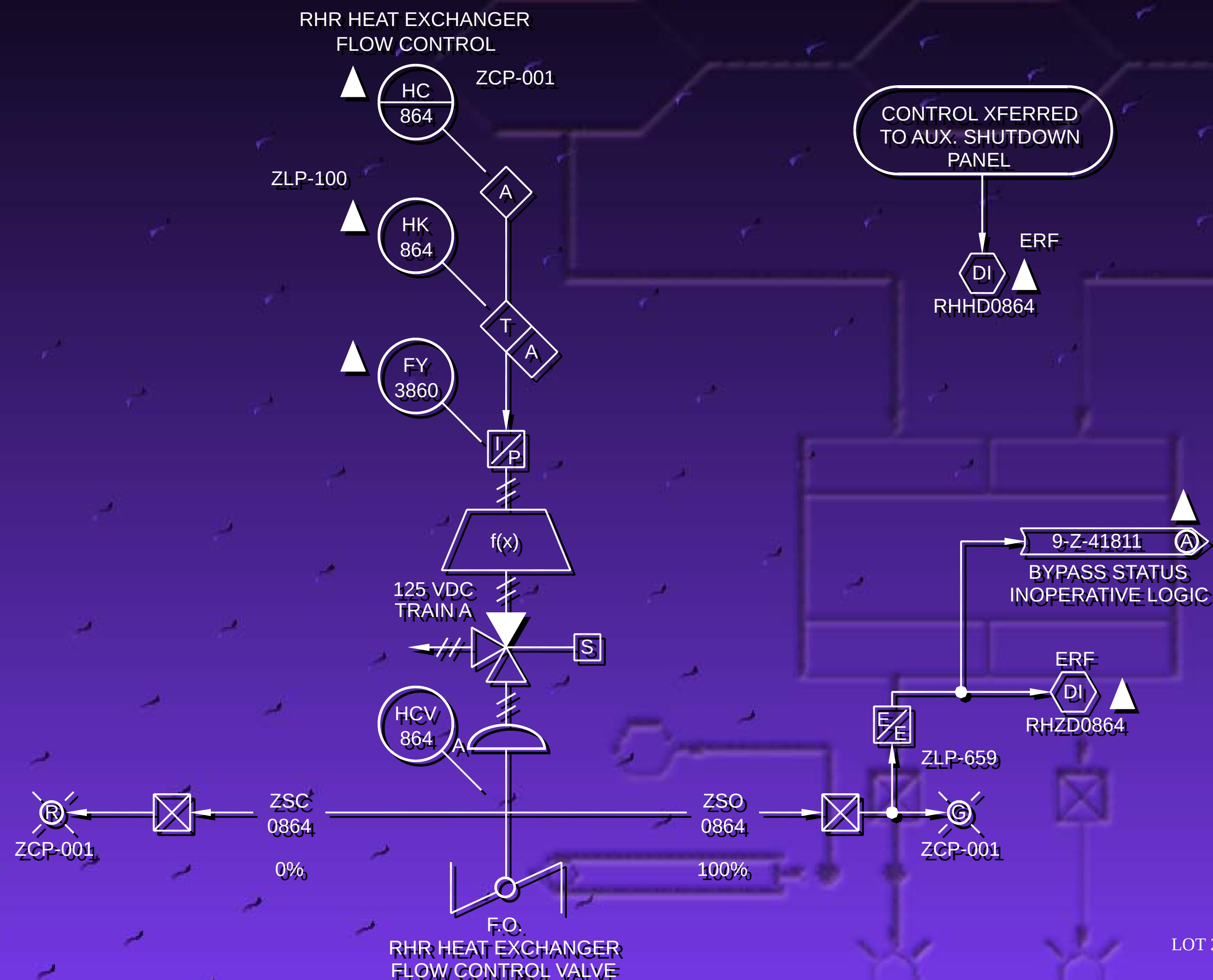
D: INCORRECT - Credible because the pumps have a large recirc valve. But unlike most pumps which have an air operated recir that fails open to protect the pump, the RHR pump recirc is motor operated (which is not affected by loss of air.

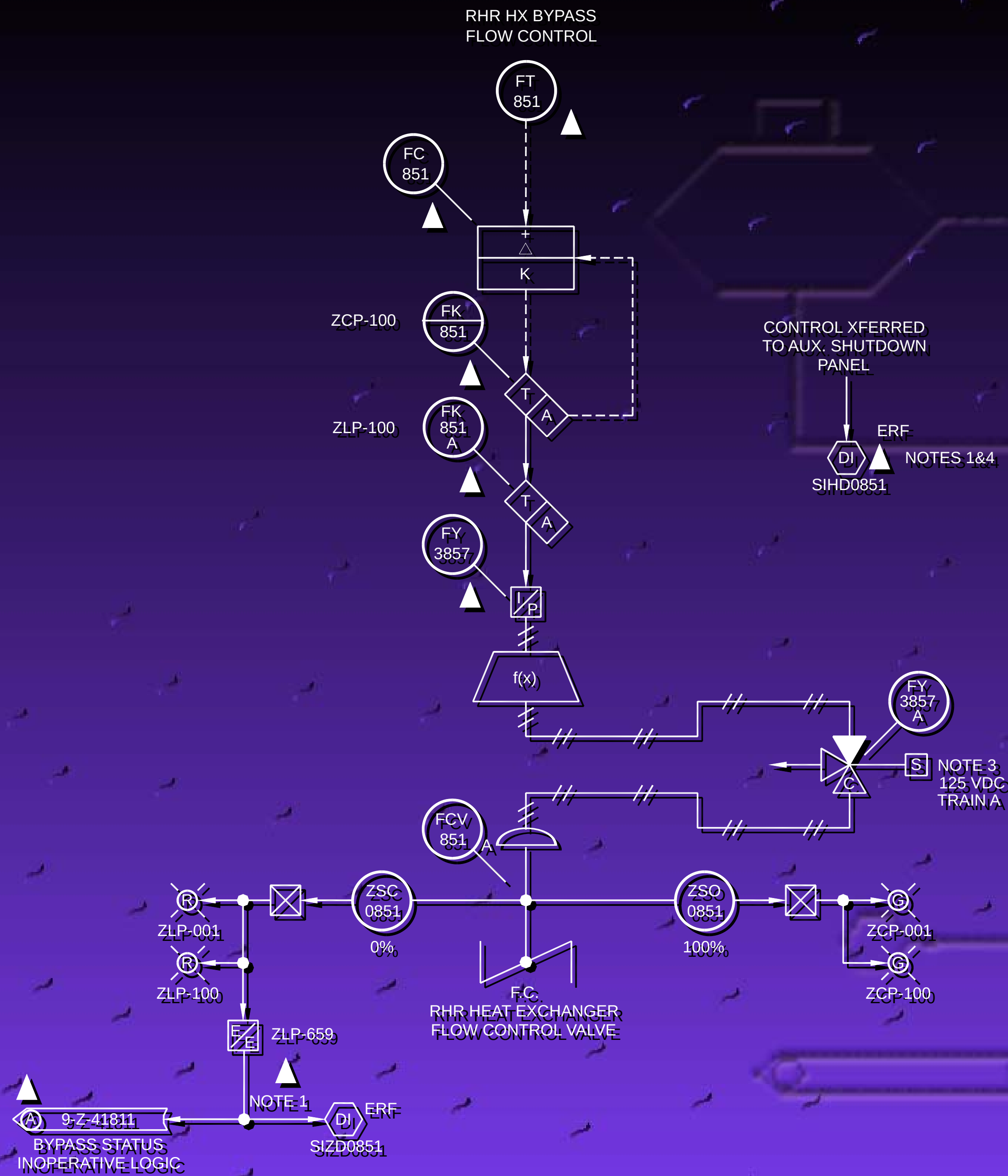
Question Level: H **Question Difficulty** 3

Justification:

The applicant must analyze the given conditions to determine the affect on the system and its availability.

RHR Heat Exchanger Flow Control Valve (9-Z-42183 Rev. 8)





RHR Heat Exchanger Bypass Control Valves (9-Z-42030 Rev. 8)

Exam Bank No.: 1559 **Last used on an NRC exam:** Never

RO Sequence Number: 45

Given the following:

- A reactor startup is in progress on Unit 1.
- SR Channel N-31 indicates 5 x 10^4 cps.
- SR Channel N-32 indicates 7 x 10^4 cps.
- IR Channel N-35 indicates 2 x 10^{-8} amps
- IR Channel N-36 indicates 2 x 10^{-10} amps

Which of the following describes the NIS response indicated by these readings?

A. All SR and IR Channels are functioning correctly; P-6 permissive is enabled.

B. SR Channel N-32 is reading abnormally high for existing conditions; P-6 permissive is NOT enabled.

C. IR Channel N-35 is reading abnormally high for existing conditions; P-6 permissive is enabled.

D. IR Channel N-36 is reading abnormally low for existing conditions; P-6 permissive is NOT enabled.

Answer: C IR Channel N-35 is reading abnormally high for existing conditions; P-6 permissive is enabled.

Exam Bank No.: 1559

K/A Catalog Number: 015 A3.03 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.9 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the NIS, including: Verification of proper functioning/operability

STP Lesson: LOT 201.16 **Objective Number:** 57121

STATE the ranges and regions of overlap for the excore NIS.

Reference: LOT201.16 PowerPoint slide 3 and 16

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because numerically, the pairs of SR and IR indications are relatively close together. The applicant must have an understanding of the meaning and significance of the indications given.

B: INCORRECT: Credible because N-32 is reading higher than N-31. The applicant must understand the significance of the the indication (numerically, the SRs are reading further apart than the IRs). P-6 is credible because 2E-8 could be construed as not greater than 1E-10 (8 is less than 10)

C: CORRECT: IR channels come on scale with SR's slightly less than 1E4 cps. IR reading of 2E-8 amps is indicative of power operation. P6 is enabled with both IR's greater than 1E-10 amps.

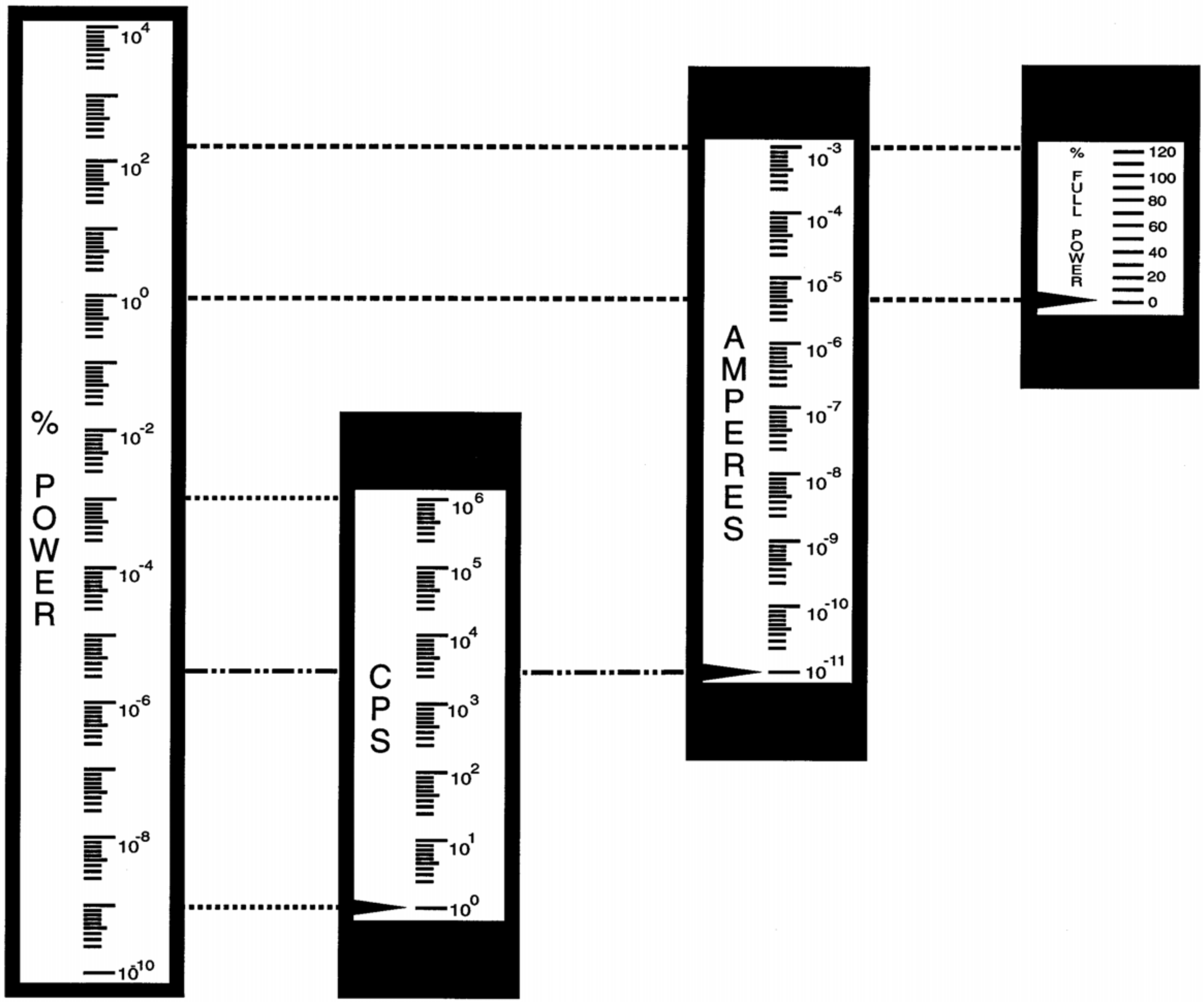
D: INCORRECT: Credible because N-36 is indicating less than N-35. N-36 is reading about where it should for the given SR indications. P6 is enabled with both IR's greater than 1E-10 amps. P-6 is credible because 2E-8 could be construed as not greater than 1E-10 (8 is less than 10)

Question Level: H **Question Difficulty** 3

Justification:

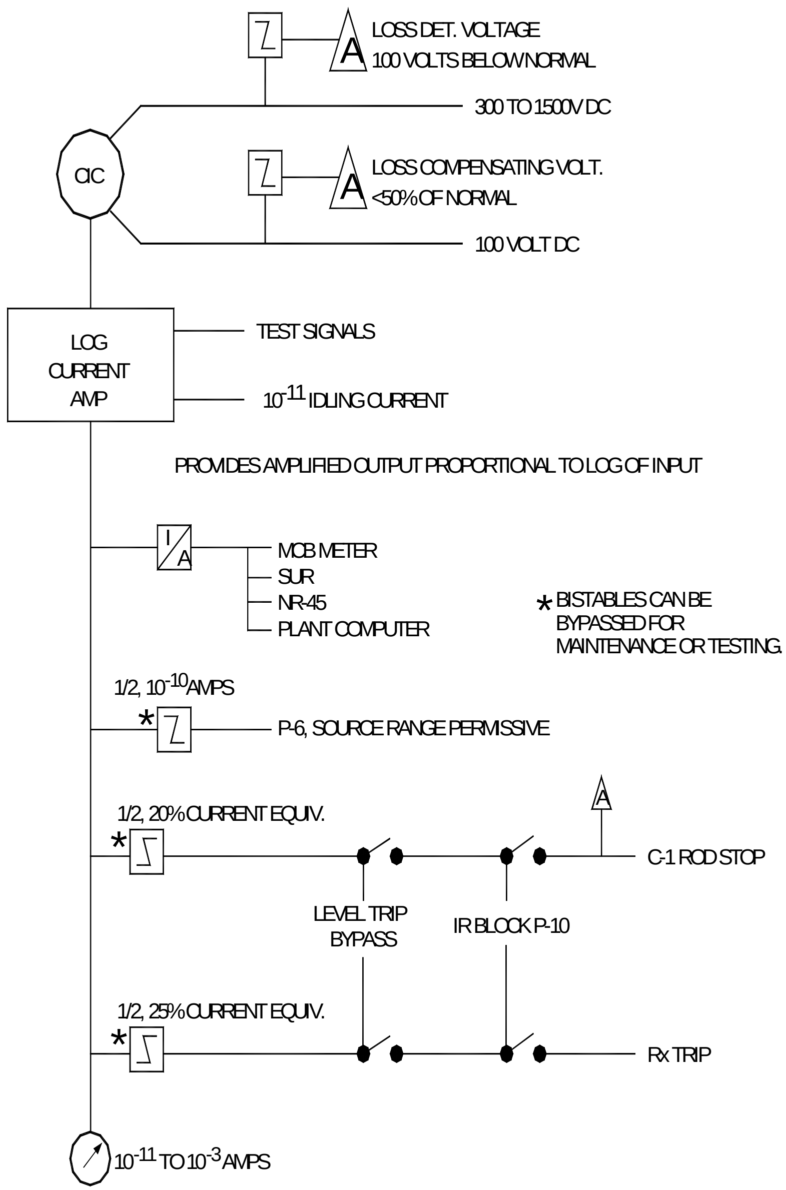
The applicant must apply knowledge of NI overlap and interlocks to the given condition to determine which reading is erroneous and P-6 status.

NUCLEAR DETECTOR RANGE OVERLAP



NLO300.39.TP.01
CDR - 07/18/96

INTERMEDIATE RANGE INSTRUMENT



LOT201.16.TP.14
DWG - 11/28/02

Exam Bank No.: 1665 **Last used on an NRC exam:** 2007

RO Sequence Number: 46

Given the following:

- Reactor power is 100%
- Control rods are in AUTO
- Channel II of Pressurizer Pressure is being calibrated and associated bistables have been TRIPPED
- Channel IV T-hot output from QDPS fails high

Which of the following describes the effect of these conditions on the Rod Control System?

A. Control rods drive in due to auctioneered Tave failed high.

B. Control rods drive in due to auctioneered ΔT failed high.

C. Reactor trip breakers open due to two channels of OPΔT bistables tripped.

D. Reactor trip breakers open due to two channels of OTΔT bistables tripped.

Answer: D Reactor trip breakers open due to two channels of OTΔT bistables tripped.

Exam Bank No.: 1665

K/A Catalog Number: 012 K3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.9 **10CFR Reference:** 55.41(b)(6)

Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS

STP Lesson: LOT 201.15 **Objective Number:** 92495

Given a description of plant conditions, PREDICT the indications received in the control room.

Reference: LOT201.15, Temperature Monitoring

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - Credible because this would be correct if the PZR pressure work was not on-going.

B: INCORRECT - Credible because RCS temperature does input into rod control (Tave rather than delta-T).

C: INCORRECT - Credible because OPΔT channel IV will trip (due to the failed T--hot), so the applicant must have knowledge of the other inputs to determine this is not correct (due to PZR pressure calibration).

D: CORRECT - Channel II OTΔT will be tripped due to the PZR Pressure channel out t of service and the Channel IV OTΔT will trip when thT-hot fails.

Question Level: H **Question Difficulty** 3

Justification:

From the given conditions, the applicant must determine the effect on the reactor protection system and the rod control system and with that knowledge determine the effect on the plant.

OTΔT Formula

- $\Delta T \leq \Delta T_o[K1 - K2(T - T') + K3(P - P') - f1(\Delta I)]$

Taverage

Measured delta T by RCS instrumentation

Pressurizer pressure

Indicated delta T at RATED THERMAL POWER

Trip stpt only reduced when delta I exceeds +8%

OPΔT Formula

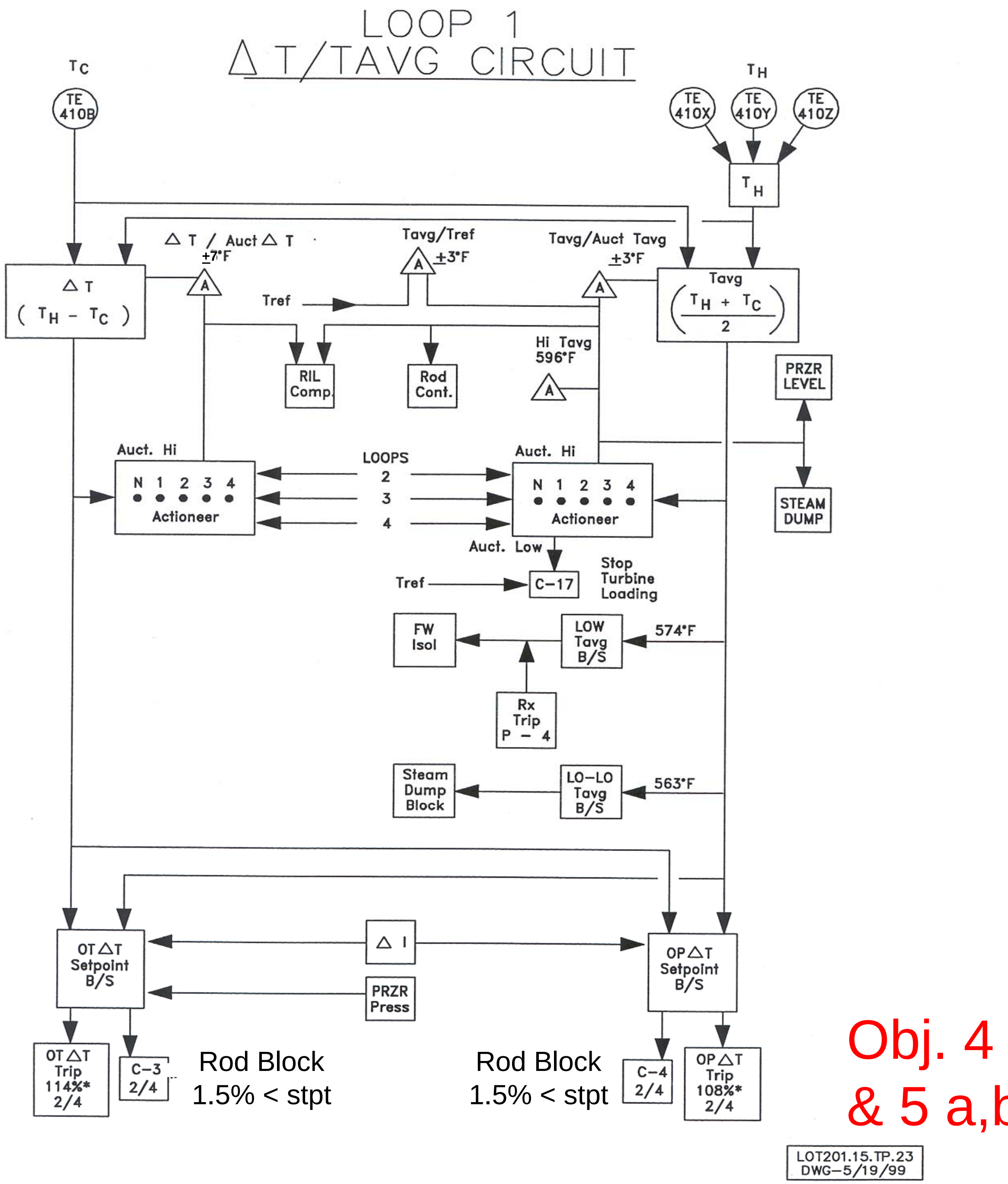
$$\Delta T \leq \Delta T_0[K4 - K5(T) - K6(T - T'') - f2(\Delta I)]$$

Taverage

Set to 0

Indicated delta T at RATED THERMAL POWER

Measured delta T by RCS instrumentation



Obj. 4 a,b,d,e,f,h & 5 a,b,c,d,f

*see COLR

Exam Bank No.: 1675 **Last used on an NRC exam:** 2007

RO Sequence Number: 47

Given the following:

- Unit 1 is in Mode 3
- Steam Dumps are in Steam Pressure Mode controlling at 1185 psig.
- All RCP's are running
- All Steam Dump Valves fail open causing the RCS to cool at >100 ºF/hr.

Which of the following correctly describes the MINIMUM operator action/s that would ensure all Steam Dumps are closed and the reason cooldown limits are established?

A. Place EITHER Steam Dump Interlock Selector Switch to 'OFF/RESET'. Excessive cooldown can result in non-ductile failure of the Reactor Vessel.

B. Place BOTH Steam Dump Interlock Selector Switches to 'OFF/RESET'. Excessive cooldown can result in ductile failure of the Reactor Vessel.

C. Place EITHER Steam Dump Interlock Selector Switch to 'BYPASS INTERLOCK'. Excessive cooldown can result in non-ductile failure of the Reactor Vessel.

D. Place BOTH Steam Dump Interlock Selector Switches to 'BYPASS INTERLOCK'. Excessive cooldown can result in ductile failure of the Reactor Vessel.

Answer: A Place EITHER Steam Dump Interlock Selector Switch to 'OFF/RESET'. Excessive cooldown can result in non-ductile failure of the Reactor Vessel.

Exam Bank No.: 1675

K/A Catalog Number: 039 K5.05 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.7 **10CFR Reference:** 55.41(b)(2)

Knowledge of the operational implications of the following concepts as they apply to the MRSS: Bases for RCS cooldown limits

STP Lesson: LOT 102.61 **Objective Number:** N99926

Describe and differentiate between the stresses induced in a reactor vessel wall during heatup and cooldown.

Reference: LOT 202.09 PowerPoint slides 101 and 103; LOT102.61 handout page 25

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT - the Steam Dump Interlock Selector Switches are redundant control devices such that either one being positioned to 'Off/Reset' will remove the control signal from all steam dumps, closing them. Cooldown limits are based on non-ductile failure (brittle).

B: INCORRECT: Switch operation is credible because some operations (i.e. CS manual actuation, blocking SI) requires operation of both train switches, however in this case both are not needed. Failure mechanism is credible because ductile failure in metals can occr, so knowledge of the difference is required.

C: INCORRECT - Switch operation is credible because 'Bypass Interlock' is another position on the Steam Dump Selector Switches, so knowledge of system operation is required.

D: INCORRECT -Switch operation is credible because 'Bypass Interlock' is another position on the Steam Dump Selector Switches, so knowledge of system operation is required. Failure mechanism is credible because ductile failure in metals can occr, so knowledge of the difference is required.

Question Level: F **Question Difficulty** 3

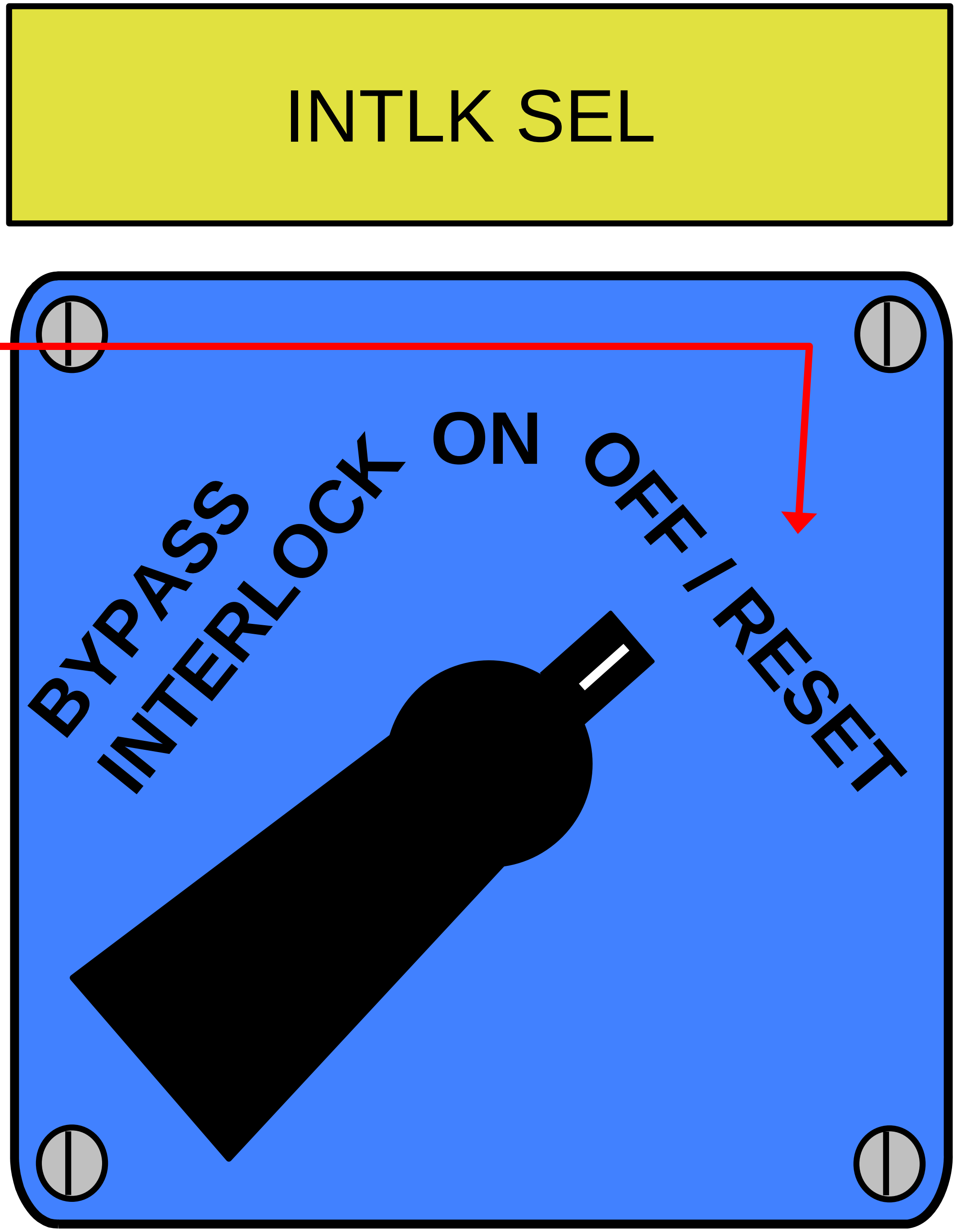
Justification:

A knowledge of Steam Dump Controls and material properties is required.

Interlock Selector Switches (Train A and B)

<u>OFF/RESET</u>

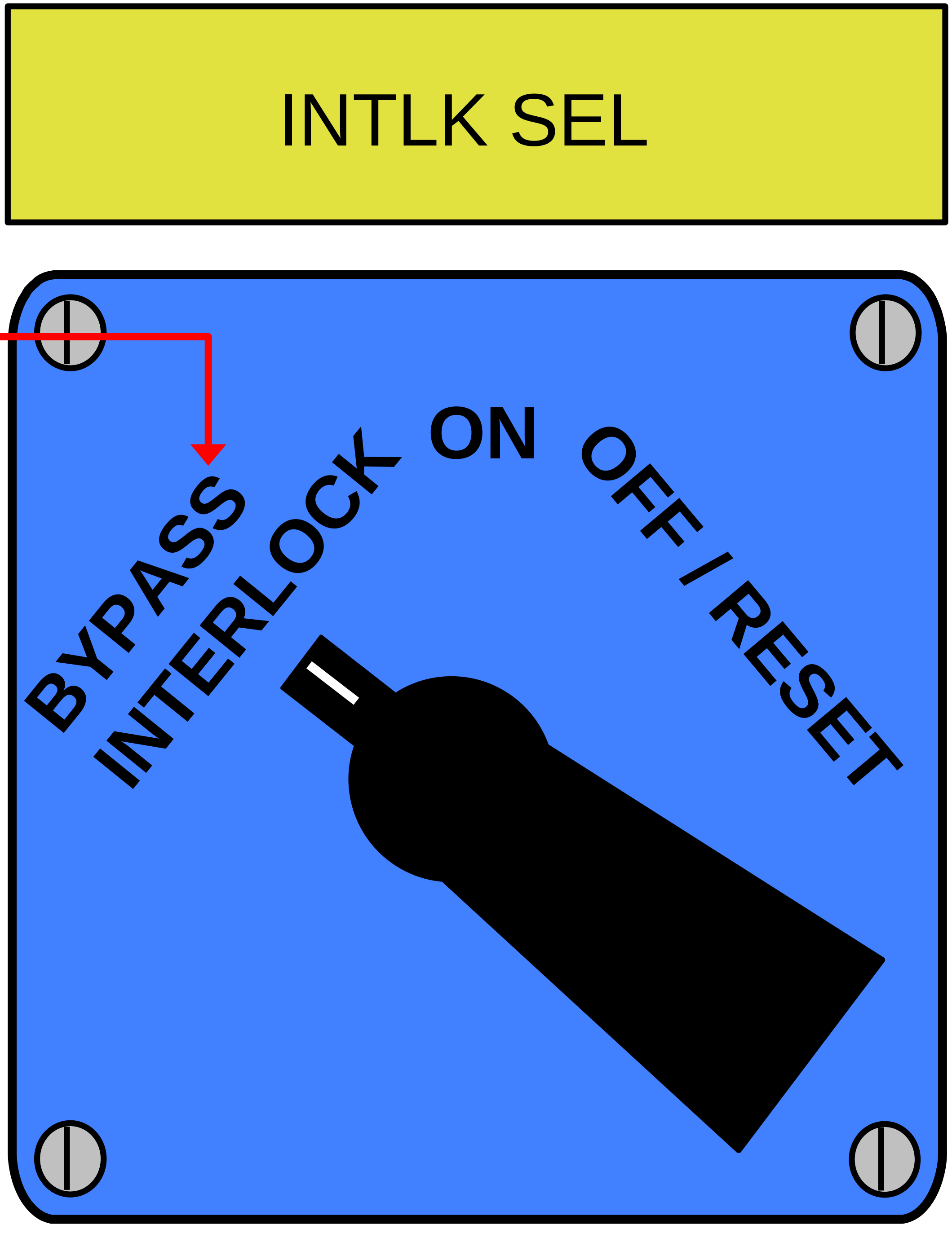
- All dump valves are blocked closed.
- After bypassing the low-low Tavg interlock for cooldown, the switch must be returned to the OFF/RESET to reactivate the interlock.



Interlock Selector Switches (Train A and B)

<u>BYPASS INTERLOCK</u>

- Spring returns to ON position.
- Allows bank No. 1 dump valves to continue dumping operations below 563°F.



For a specific irradiated nuclear pressure vessel, the embrittlement depends on the neutron doses, neutron spectrum, irradiation temperature, steel material, and the amount of trace impurities, copper and phosphorous. For trace impurities, shifts in transition temperature by as much as 100°F above the shift otherwise predicted have been observed for increases in copper content from ≈ 0.1% to ≈ 0.2% at a flux of ≈ 2×10^{19} n/cm^2. Thus the change in transition temperature is quite sensitive to impurity level.

REACTOR VESSEL STRESSES

To prevent brittle fracture, a vessel must not be stressed too heavily while it is cool. As its temperature increases, it can withstand higher pressures since the metal becomes more ductile. Engineers can calculate the minimum metal temperature required to prevent brittle fracture, given the stress conditions at a certain pressure and heatup rate, as shown in Figure 10-8.

It is possible to construct curves showing the minimum required wall temperature to prevent failure at a given reactor pressure and heatup rate. This would not be of much use to control room operators, however, as they have no direct indication of vessel wall temperatures.

To circumvent this problem, the engineers took into account the fact that the metal temperature would lag the coolant temperature as the coolant temperature rose or fell. Once the temperature differences had been calculated, the minimum reactor temperatures required to safely support the existing pressures were computed. This information is useful to operators because they have direct indication of coolant pressures and temperatures. These parameters are used to establish and control vessel heatup and cooldown rate.

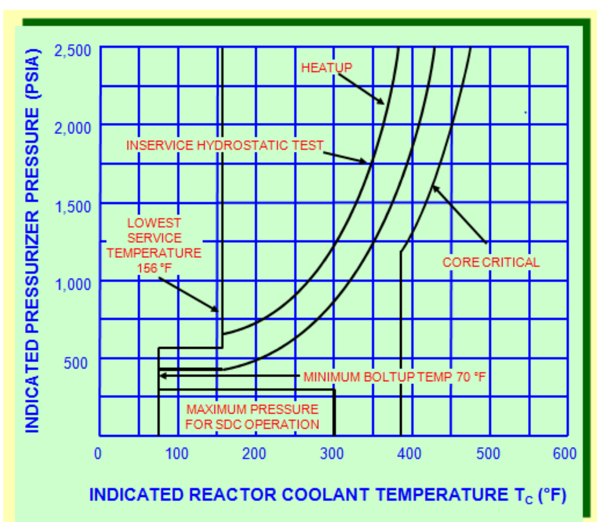


Figure 10-8 Minimum Vessel Temperature vs. Vessel Pressure

The reactor vessel and associated piping are normally pressurized from atmospheric pressure up to greater than 2,000 psig. The stress from the pressure is called tensile stress. Both the inner and outer walls of the vessel are subject to tensile stress, with the inner wall experiencing the greatest stress.

Exam Bank No.: 1700 **Last used on an NRC exam:** Never

RO Sequence Number: 48

Which of the following correctly describes a condition that could cause the DRPI and the step counter for a particular control rod group to disagree AND result in a Tech Spec entry?

A. Rod Control Logic Cabinet Urgent Alarm.

B. Rod Control Logic Cabinet Non-Urgent Alarm

C. DRPI Urgent Alarm

D. DRPI Non-Urgent Alarm

Answer: C DRPI Urgent Alarm

Exam Bank No.: 1700

K/A Catalog Number: 014 K5.01 **Tier:** 2 **Group/Category:** 2

RO Importance: 2.7 **10CFR Reference:** 55.41(b)(5)

Knowledge of the operational implications of the following concepts as they apply to the RPIS: Reasons for differences between RPIS and step counters.

STP Lesson: LOT 201.19 **Objective Number:** 98055

GIVEN a plant or system condition, PREDICT the operation of the Rod Position Indication System.

Reference: LOT 201.19 PowerPoint slide 27

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because a logic cabinet urgent failure will prevent rods from moving in manual or auto and result in a Tech Spec entry, however DRPI and step counters will still agree.

B: INCORRECT: Credible because it is a logic cabinet alarm, and without in-depth knowledge of the system could be construed as a possible cause.

C: CORRECT: a DRPI urgent failure will cause a loss of DRPI indication for that rod, thus there will be disagreement with the step counter position for that rod. Would enter the TS for an inoperable DRPI channel.

D: INCORRECT: Credible because it is a DRPI alarm and could cause a difference in the board indication (but not Tech Spec entry).

Question Level: H **Question Difficulty** 3

Justification:

The applicant must have a knowledge of the urgent and non-urgent alarms involved and apply it to the circumstances described in the question. A basic knowledge of Tech Spec requirements is also needed.

ALARMS AND INDICATIONS

DISPLAY UNIT

ALARMS

URGENT NON-URGENT DEVIATION

URGENT ALARM

Data A and B Failure

Detector Failure

Control Unit Failure

Exam Bank No.: 1726 **Last used on an NRC exam:** Never

RO Sequence Number: 49

If the fire detectors in the EAB 35' Relay Room do not function, then the __________ system in the Relay Room ________________.

A. Carbon Dioxide (CO_2); can still be actuated manually

B. Halon; can still be actuated manually

C. Halon; will not actuate manually or automatically

D. Carbon Dioxide (CO_2); will not actuate manually or automatically

Answer: B Halon; can still be actuated manually

Exam Bank No.: 1726

K/A Catalog Number: 086 K6.04 **Tier:** 2 **Group/Category:** 2

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(7)

Knowledge of the effect of a loss or malfunction of the following will have on the Fire Protection System: Fire, Smoke, and Heat Detectors.

STP Lesson: LOT 201.29 **Objective Number:** 91394

LIST the types of fire detectors used and DESCRIBE their basic principle of operation.

Reference: LOT201.29, lesson on Fire Protection

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because CO2 extinguishers are available in the area, but the relay room uses Halon for fire protection.

B: CORRECT: The relay room uses Halon for fire protection. Manual actuation is always available.

C: INCORRECT: Credible because the use of detectors and auto or manual actions would prevent inadvertent operation of the system.

D: INCORRECT: System type is credible because CO2 extinguishers are available in the area, but the relay room uses Halon for fire protection. Action is credible because the use of detectors and auto or manual actions would prevent inadvertent operation of the system.

Question Level: F **Question Difficulty** 2

Justification:

The applicant requires a basic knowledge of the fire protection system located in the Relay Room.

G. Foam-Water Sprinkler Systems (3)

This system is installed to protect the Standby Diesel Generator Fuel Oil Storage Tanks (FOST) located above the engine rooms. Each of the Foam-Water Sprinkler Systems is dedicated to protect only one diesel fuel tank. The system operation is basically the same as described above for the AFOST with the exception that instead of sending foam solution through a nozzle, it passes through a sprinkler designed to mix air with the solution to produce a foam spray.

The water supply to the foam-water system is supplied from the Ring Main by deluge valves in separate valve houses for each diesel generator located adjacent to the north wall of the DGB. The foam concentrate for each of the three systems is stored in a separate 100 gallon capacity tank located next to the respective FOST room.

H. Halon 1301 System

Halon systems consist of a stored pressurized gas that, when released, stops the combustion reaction of fire by chemical interaction. Halon is the trade name for halogenated agent bromotrifluoromethane and is much safer for use in fire suppression than other gases such as Carbon Dioxide because it is breathable in the range of 7 to 7.5% concentration for which the system is designed.

The process computer room (including the raised floor), computer battery room, and relay room on the 35 foot elevation and the Technical Support Center (TSC) on the 70 foot elevation of the EAB are protected by halon systems. The halon system supplying the rooms on the 35 foot elevation is located on the 10 foot elevation of the EAB inside a room adjacent to the corridor on the west end. The TSC halon system is separate and located on the 70 foot elevation.

The Halon System that protects the EAB computer and relay rooms consists of 20 storage bottles for the main bank and 20 in a reserve bank. The pressurized storage bottles and a Halon fire control panel is located in the room on the 10 foot elevation of the EAB previously mentioned. This system uses halon bottles pressurized to 600 psig and is designed to flood the computer or relay room with a concentration of 7 to 7.5% within 10 seconds of receiving the fire alarm and actuation signal.

Automatic actuation of the system is initiated by the Halon fire control panel when 1 of 2 ionization smoke detectors in each of two redundant (cross) zones of detection sense a fire in that area. The use of cross zoning prevents an inadvertent halon actuation should a detector fail. When the halon system is actuated, the affected area ventilation dampers are automatically closed, and solenoid valves open on the selected halon cylinders and main supply line to flood the area with halon. A pre-discharge evacuation alarm is sounded in the affected area to warn personnel. 20 bottles are released for a relay room fire and 5 are used for a computer room fire.

Manual fire alarm switches are provided at the entrance to each room to actuate the system. The system can also be actuated by mechanical dump at the cylinders. A selector switch is located in the halon system room to allow selection of the reserve bank after the main bank has been used. A pressure switch is located in the halon system discharge piping to warn operators that a halon system actuation has occurred.

I. Portable Fire Extinguishers

Portable Extinguishers are available throughout the plant. Carbon Dioxide, pressurized water, and dry chemical extinguishers are provided in selected locations based on their anticipated needs.

1.3 General Design Criteria

1.3.1 The Fire Protection System is designed in accordance with 10 CFR 50 and other industrial requirements. The following are those applicable to this lesson:

- Safety-related structures that have exposed steel are protected with spray-applied fire proofing material that has a fire rating of at least three hours.
- Materials of low combustible and/or low fuel contribution, flame spread, and smoke development are used within safety-related structures.
- An automatic sprinkler system is provided for the one case where open cable trays are routed above a suspended ceiling in the Health Physics Office area in the MAB.
- Floor drains which may collect water from a radioactive area are routed to the LWPS.
- Floor drains are designed to prevent fire spread from one drainage area to another.

Exam Bank No.: 2167 **Last used on an NRC exam:** Never

RO Sequence Number: 50

You 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', on the ESF Status Panel, you note the following 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 correctly describes the status of Phase 'A' Isolation and any actions that may be required?

A. At least one Train 'A' valve is open; manually close the valve(s).

B. At least one Train 'B' valve is open; manually close the valve(s).

C. At least one Train 'A' valve is open; manually actuate Phase 'A' isolation.

D. At least one Train 'B' valve is open; manually actuate Phase 'A' isolation.

Answer: A At least one Train 'A' valve is open; manually close the valve(s).

Exam Bank No.: 2167

K/A Catalog Number: 103 A3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.9 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the containment system, including: Containment isolation

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: 0POP05-EO-EO00 Rev 22

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT - The red lights tell the operator all trains received an actuation signal. The F/ACT light tells the operator the status of the actuation. If the light is lit, then the actuation is not fully complete, as is the case for train 'A'.

B: INCORRECT - Credible because the BYP INOP light is lit which tells the operator that component conditions could prevent an actuation from occuring, but the fact that the F/ACT light is off for the train indicates the actuation is complete.

C: INCORRECT - Credible because if an actuation has not occurred, then a manual action would be required. In this case, the actuation signal is present, but all components did not actuate so a manual action will not help.

D: INCORRECT - Status is credible because the BYP INOP light is lit which tells the operator that component conditions could prevent an actuation from occuring, but the fact that the F/ACT light is off for the train indicates the actuation is complete. Action is credible because if an actuation has not occurred, then a manual action would be required. In this case, the actuation signal is present, but all components did not actuate so a manual action will not help.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must first determine the meaning of the light indication given in the stem. Then using their knowledge of sytem design, determine the correct course of action for the condition given.

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	<u>WHEN</u> the respective ESF Load Sequencer has completed its automatic sequence <u>OR</u> it is determined that the respective ESF Load Sequencer has failed, <u>THEN</u> PERFORM the following: a. Manually START pump(s). b. Manually OPEN discharge isolation valve(s). c. <u>IF</u> any ECW pump can <u>NOT</u> be started <u>OR</u> its discharge isolation valve can <u>NOT</u> be opened, <u>THEN</u>: 1) STOP associated STBY DG by PLACING in PULL-TO-STOP. 2) ENSURE associated Essential Chiller(s) TRIP.
___ 8	VERIFY CCW pumps - RUNNING	<u>WHEN</u> the respective ESF Load Sequencer has completed its automatic sequence <u>OR</u> it is determined that the respective ESF Load Sequencer has failed, <u>THEN</u> manually START pump(s).

Exam Bank No.: 2175 **Last used on an NRC exam:** Never

RO Sequence Number: 51

Unit 1 is in MODE 4 with a plant heat up in progress.

- A Train CCW is out for maintenance.
- Train 'B' and 'C' RHR are in service.
- RCS Temperature is 345°F.

Subsequently one of the suction valves closes on Train 'B' RHR.

Which of the following describes (1) the impact of this malfunction to Train 'B' RHR and (2) the actions that should be taken to prevent an inadvertent entry in to MODE 3?

A. (1) Train 'B' RHR trips on low flow.
(2) Use SG PORVs to steam SGs per 0POP03-ZG-0001, Plant Heatup.

B. (1) Train 'B' RHR trips on low flow.
(2) Start Train 'A' RHR per 0POP02-RH-0001, Residual Heat Removal System Operation.

C. (1) Train 'B' RHR miniflow valve automatically opens due to low flow.
(2) Secure Train 'B' RHR and Start Train 'A' RHR per 0POP02-RH-0001, Residual Heat Removal System Operation.

D. (1) Train 'B' RHR miniflow valve automatically opens due to low flow.
(2) Secure Train 'B' RHR and use SG PORVs to steam SGs per 0POP03-ZG-0001, Plant Heatup.

Answer: A (1) Train 'B' RHR trips on low flow.
(2) Use SG PORVs to steam SGs per 0POP03-ZG-0001, Plant Heatup.

Exam Bank No.: 2175

K/A Catalog Number: 05 A2.04 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(3)

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 those malfunctions: RHR valve malfunction

STP Lesson: LOT 201.09 **Objective Number:** 4245

Given a plant or system condition, predict the operation to the Residual Heat Removal System.

Reference: 0POP02-RH-0001, RHR System Operation, discusses the RHR low flow trip and 0POP03-ZG-0001, Plant Heatup, discusses use of SG PORVs prior to and when transitioning form MODE 4 to MODE 3.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: RHR Pumps will trip on low flow and SGs can be used to control RCS temperature when at the transition point for going from MODE 4 to MODE 5.

B: INCORRECT: Credible since under normal conditions, this could be correct. However it is given that CCW 'A' is OOS, so RHR 'A' would not be placed in service.

C: INCORRECT: Impact is credible because most recirc valves will automatically maintain minimum flow for the pump, however the RHR pump recircs have no auto function. Action is credible since under normal conditions, this could be correct. However it is given that CCW 'A' is OOS, so RHR 'A' would not be placed in service..

D: INCORRECT: Impact is credible because most recirc valves will automatically maintain minimum flow for the pump, however the RHR pump recircs have no auto function.

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator must evaluate the given conditions to determine the effects of the malfunction and the actions to take.

Initials

7.0 Mode 3 Heat-Up

CAUTION

Cold Shutdown (68°F, Xenon-free) RCS boron concentration SHALL be maintained while SI Actuation below P-11(1985 psig) is Blocked. (Reference 2.5.35 & 2.5.43)

7.1 IF Core Exit Thermocouple/Resistance Temperature Detector Cross Calibration is to be performed, THEN REVIEW 0PSP10-RC-0002 "Core Exit Thermocouple/Resistance Temperature Detector Cross Calibration" Section 5.3 for plant parameters necessary for the TEST. _____

7.2 PLACE a minimum of two CRDM vent fans and one reactor cavity vent supply and exhaust train in operation per 0POP02-HC-0001, Containment HVAC. _____

NOTE

WHEN RCS temperature is greater than 340°F, THEN the S/U SGFP 14(24) is the preferred method to feed the Steam Generators.

7.3 WHEN RCS temperature is greater than 340°F, THEN PERFORM the following to feed the SGs.

7.3.1 IF S/U SGFP 14(24) is **NOT** available AND Feedwater Booster Pump desired for feed, THEN GO TO Step 7.3.3.1. _____

7.3.2 IF S/U SGFP 14(24) is **NOT** available, THEN CONTINUE AFW operation. _____

7.3.3 IF S/U SGFP 14(24) is available, THEN PERFORM the following:

7.3.3.1 ESTABLISH feedwater flow to the SGs using S/U SGFP 14(24) OR Feedwater Booster Pump per 0POP03-ZG-0003, Secondary Plant Startup. _____

7.3.3.2 SECURE all AFW pumps per 0POP02-AF-0001, Auxiliary Feedwater. _____

7.3.3.3 ENSURE the ESF Standby Readiness Lineup Section 16.0 has been performed per 0POP02-AF-0001, Auxiliary Feedwater. _____

Initials

CAUTION

The Main Steam lines upstream of the MSIVs may require periodic blowdown for moisture control. This can be accomplished by performing Addendum 5. (CR 03-3694)

7.3.3.4 INITIATE warmup of the Main Steam lines per 0POP03-ZG-0003, Secondary Plant Startup, or 0POP03-ZG-0011, Secondary Plant Cold Startup, per the Unit Operations Manager. _____

NOTE

- Either steaming method mentioned in Steps 7.4 and 7.5 may be used.
- IF SG PORVs are required to be controlled in manual to maintain RCS or Secondary Side temperatures, THEN an OAS entry is required to ENSURE compliance with TS 3.3.5.1 and TS 3.7.1.6.
- IF RHR was REMOVED from service and RCS Pressure is between 550 psig and 600 psig in from Step 6.22, THEN Steps 7.7, 7.8 and 7.9 are NA.
- IF the (Alternate) steam dumps method is used in Step 7.4, THEN monitor hotwell levels to prevent overfill.

7.4 IF SGs are being fed with AFW, THEN ESTABLISH RCS cooling by steaming SGs utilizing the (Preferred) SG PORVs or (Alternate) steam dumps. {CP006} _____

7.5 IF SGs are being fed with Main Feedwater, THEN ESTABLISH RCS cooling by steaming SGs utilizing steam dumps or SG PORVs. {CP007} _____

7.6 MAINTAIN SG Narrow Range levels between 55 and 75%. _____

7.7 IF RHR is in service, THEN REDUCE cooling by the RHR system as heat removal is established by means of the SGs. _____

7.8 ENSURE RHR system REMOVED from operation per 0POP02-RH-0001, Residual Heat Removal System Operation. _____

7.9 WHEN RHR has been removed from service, THEN RAISE RCS Pressure to between 550 psig and 600 psig. _____

7.10 ESTABLISH required ECCS lineup per 0POP02-RH-0001, Residual Heat Removal System Operation. _____

Exam Bank No.: 2176 **Last used on an NRC exam:** Never

RO Sequence Number: 52

Unit 2 is in MODE 3 with all 4 Reactor Coolant Pumps (RCP) operating, when a breaker fault causes RCP 2A to trip.

Which of the following describes the response of the LOOP A FLOW indicator?

Indication …

A. instantly drops to 0% and stabilizes

B. instantly drops to 0%, then rises to ~20%

C. drops to 0% over a period of time and stabilizes

D. drops to 0% over a period of time, then rises to ~20%

Answer: D drops to 0% over a period of time, then rises to ~20%

Exam Bank No.: 2176

K/A Catalog Number: 003 K5.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.8 **10CFR Reference:** 55.41(b)(5)

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP coastdown on RCS parameters

STP Lesson: LOT 201.05 **Objective Number:** 86369

DESCRIBE the effects on the plant due to tripping a Reactor Coolant Pump.

Reference: LOT201.05 handout page 7

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because pumps without flywheels (most pumps) will exhibit an immediate drop in flow when secured. Also other systems with multiple pumps do not have reverse flow in the idle pumps, so it is not expected to have indicated flow on an idle pump.

B: INCORRECT: Credible because pumps without flywheels (most pumps) will exhibit an immediate drop in flow when secured.

C: INCORRECT: Credible because other systems with multiple pumps do not have reverse flow in the idle pumps, so it is not expected to have indicated flow on an idle pump.

D: CORRECT: The flywheel is designed to maintain flow for a short period of time following a trip of the pump. Once the pump stops, the core DP generaterd by the 3 running pumps will force a small amount of flow backwards through the idle loop and cause the flow indicator to rise again.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must understand the design features of the flywheel and then must realize that with the other 3 pumps running there will be reverse flow through the idle loop due to the reactor vessel DP.

The flywheel stores inertial energy to keep the pump rotating for a short but critical period following station blackout. The flywheel provides approximately one minute of flow coastdown to provide DNB protection during the early stages of station blackout when the decay heat level is high. The flywheel is keyed to the motor shaft above the upper bearing assembly.

In a multi-loop plant, de-energization of one or more reactor coolant pumps while another pump or pumps are running causes a reverse flow through the inactive loops. This reverse flow tends to turn the de-energized pumps backwards. Although no mechanical damage would result from such reverse rotation, if an attempt were made to start a pump in this condition, excessive starting currents would be drawn for an excessive time, resulting in over-heating of the motor. To prevent this reverse rotation, each pump is equipped with an anti-reverse rotation device. The anti-reverse mechanism consists of five pawls mounted on the outside diameter of the flywheel, a serrated ratchet plate mounted on the motor frame, a spring return for the ratchet plate, and two shock absorbers. After the motor has come to a stop, one pawl will engage the ratchet plate and, as the motor starts to rotate in the opposite direction, the ratchet plate will also rotate slightly until stopped by the shock absorbers. The rotor will remain in this position until the motor is energized again. After the motor has started to rotate, the ratchet plate will be returned to its original position by the spring return. When the motor is started, the pawls will drag over the ratchet plate until the motor reaches approximately 70 rpm. After this time, centrifugal force will keep the pawls in an elevated position.

STATOR

The stator core laminations are made from high-silicon electrical sheets coated with alkoplus for insulation. Stacks of laminations periodically separated by air vent spacers are held in place by studs and clamped by steel end plates.

The stator windings are made of insulated copper wire fitted into slots in the core. The ends of the windings, which extend beyond the slots, are braced by an insulated support ring and are separated by non-woven polyester felt packing to withstand the mechanical forces associated with full-voltage starts.

The entire stator core and windings are insulated and are moisture resistant. A solid epoxy resin strengthens the bracing system of the windings. This resin is susceptible to radiation damage at levels of 100 rads/hr. Normal radiation levels are less than 50 rads/hr; therefore, no deterioration is expected.

RCP ELECTRICAL TRIPS

Trips for the RCPs are:
Undervoltage – sensed on the 13.8 Kv Aux Bus cub 13
Reverse Phase – sensed on the 13.8 Kv RCP breaker (Aux bus cubicle 11)
Overcurrent - sensed on the 13.8 Kv RCP breaker (Aux bus cubicle 11)
Underfrequency – sensed on the Class 1E 15 Kv RCP Cubicle (Dummy breaker). This signal feeds into SSPS and on a 2 of 4 logic, trips ALL RCPS and also generates a Reactor Trip.

Exam Bank No.: 2179 **Last used on an NRC exam:** Never

RO Sequence Number: 53

Given the following:

- Pressurizer pressure is 2235 psig
- PRT PRESS HI annunciator is in
- PRT TEMP HI annunciator is in
- PRT pressure on CP-04 is indicating 10 psig and slowly rising
- PRT temperature on CP-04 is indicating 110 F and slowly rising

Which of the following is the likely cause for this condition?

A. Pressurizer PORV seat leakage.

B. Reactor Makeup Water to the PRT (FV-3650) is open.

C. Letdown Stop valve leakoff flow high.

D. RCP #1 seal leakoff flow high.

Answer: A Pressurizer PORV seat leakage

Exam Bank No.: 2179

K/A Catalog Number: 007 A4.10 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(3)

Ability to manually operate and/or monitor in the control room: Recognition of leaking PORV/code safety

STP Lesson: LOT 201.04 **Objective Number:** 80883

DESCRIBE the indications available to determine that a Pressurizer power operated relief valve is leaking.

Reference: LOT201.04 PZR, PRT and RCDT Power Point Presentation Slide #58

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: The discharge of the PZR PORV is directed to the PRT this would cause temperature and pressure to rise in the PRT.

B: INCORRECT: Credible because this flowpath would cause PRT pressure to rise, but temperature should not.

C: INCORRECT: Credible because these indications would occur in the RCDT (another collection tank located in containment). The applicant must have knowledge of the different system flowpaths.

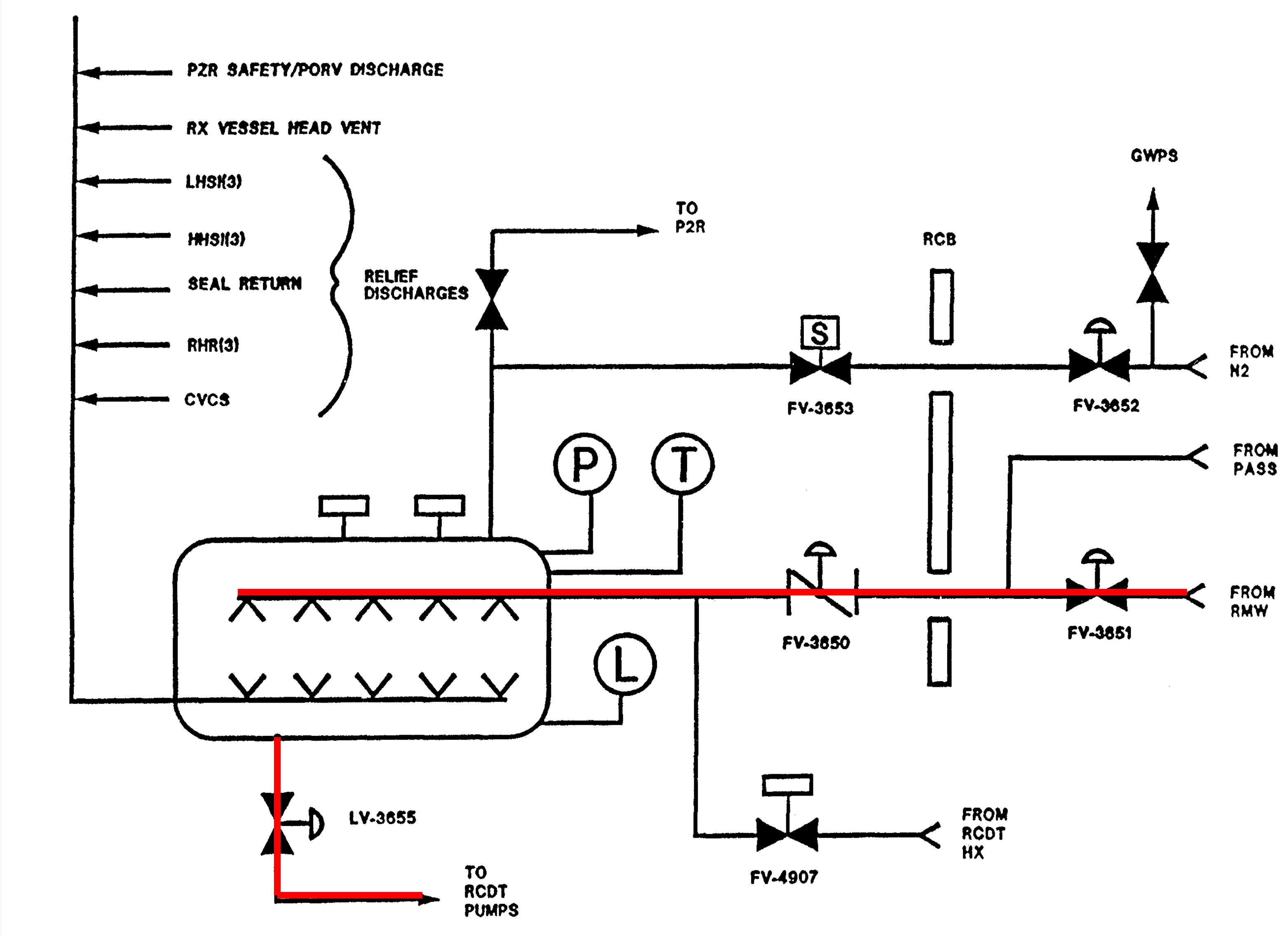
D: INCORRECT: Credible because this source would give these indications in the PRT following an SI actuation, but not during normal operations. There is no indication in the stem that an SI has occurred.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must recognize from indications that a high energy source is going into PRT and recall which influents are directed to PRT and which are high enough energy to give the indications provided.

PRT FEED AND BLEED



Exam Bank No.: 2186 **Last used on an NRC exam:** Never

RO Sequence Number: 54

Given the following:

- Unit 1 is at 100% power
- 'A' Train ECW/CCW systems are running
- 'B' Train ECW is running
- 'B' ECW/CCW is the Train selected for STBY.
- 'A' CCW HX outlet flow indicates 12,700 gpm

Subsequently, the following occurs:

- CCW HX 1A 'OUTL FLOW HI/LO' alarms
- CCW HX 1A 'OUTL PRESS LO' alarms
- CCW HX 1A OUTL PRESS PI-4513 indicates 74 psig

Select the malfunction that could have caused the given indications and the automatic action that should occur.

	MALFUNCTION	AUTOMATIC ACTION
A.	A loss of power to 'RHR 1A CCW Outlet Valve' caused the valve to fail open.	The 1B CCW Pump will auto start after a time delay.
B.	A loss of power to 'RHR 1A CCW Outlet Valve' caused the valve to fail open.	The 1B CCW Pump will start immediately.
C.	A loss of power to 'CCW HX 1A Outlet TCV' caused the valve to fail open.	The 1B CCW Pump will auto start after a time delay.
D.	A loss of power to 'CCW HX 1A Outlet TCV' caused the valve to fail open.	The 1B CCW Pump will start immediately.

Answer: A A loss of power to 'RHR 1A CCW Outlet Valve' caused the valve to fail open The 1B CCW Pump will auto start after a time delay.

Exam Bank No.: 2186

K/A Catalog Number: 008 A3.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.0 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the CCWS including:
All flow rate indications and the ability to evaluate the performance of this closed cycle cooling system.

STP Lesson: LOT 201.12 **Objective Number:** 57126

Describe the operation of the Component Cooling Water System and its major components. Include automatic actions, interlocks and trips.

Reference: LOT 201.12 Power Point Presentation Slide #8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: The RHR HX CCW Oulet Valve fails open on loss of power or loss of air. The train flow will increase by approximately 5000 gpm, Stby CCW pump auto starts on low common header pressure of 76 psig after a time delay.

B: INCORRECT: Pump start is credible because that is original plant design and the way most systems function.

C: INCORRECT: Valve failure is credible because open failure of a safety related TCV would be reasonable and desired, however these valves are MOVs and will fail as is if power is lost to the valve.

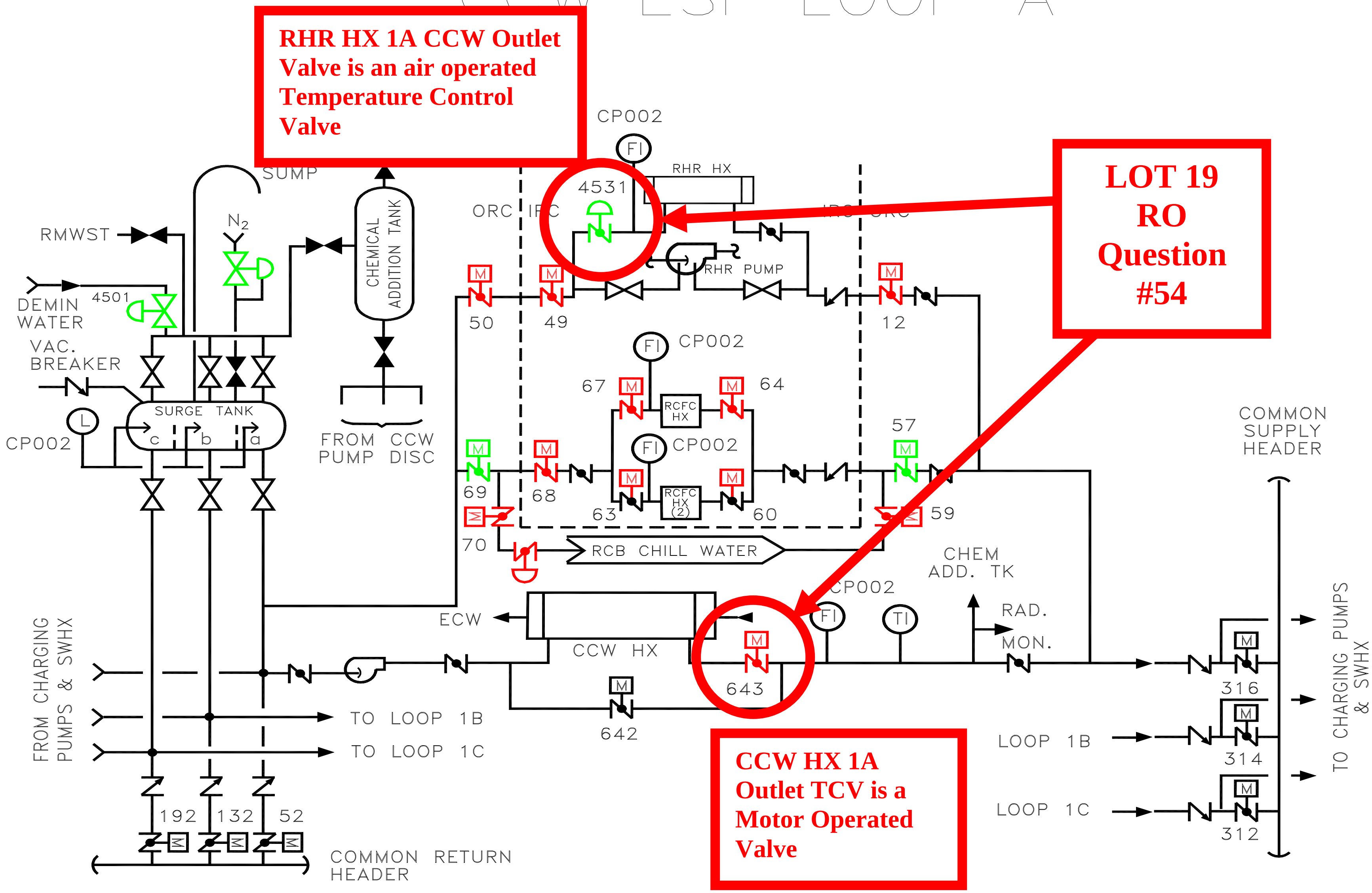
D: INCORRECT: Valve failure is credible because open failure of a safety related TCV would be reasonable and desired, however these valves are MOVs and will fail as is if power is lost to the valve. Pump start is credible because that is original plant design and the way most systems function.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must recognize that the RHR HX outlet valve has failed open given lost valve indication, high flow alarm and low system press. And that the stby train will auto start after a time delay.

CCW ESF LOOP A



Exam Bank No.: 2189 **Last used on an NRC exam:** Never

RO Sequence Number: 55

Given the following:

- Unit 1 is at 100% Power.
- Train 'A' and 'C' ECW/CCW pumps are running to support Unit operations
- Train 'B' ECW/CCW is in standby.

Subsequently the following occur simultaneously:

- A large Steam Line Break in Containment
- MCC E1B3 loses power.

Which of the following describes an impact due to these events and the actions that will mitigate the consequences?

	IMPACT	ACTIONS
A.	CS Pump 1B discharge valve is CLOSED and will not OPEN.	• Place CS Pump 1B in PTL to prevent running the pump at shutoff head. • Start two additional RCFCs to provide additional containment cooling.
B.	CS Pump 1B discharge valve is OPEN and will not CLOSE.	• Place CS Pump 1B in PTL to prevent runout conditions. • Start two additional RCFCs to provide additional containment cooling.
C.	ECW Pump 1B discharge valve is CLOSED and will not OPEN.	• Place ECW Pump 1B in PTL to prevent running the pump at shutoff head. • Place EDG #12 in PTL to prevent overheating.
D.	ECW Pump 1B discharge valve is OPEN and will not CLOSE.	• Place ECW Pump 1B in PTL to prevent runout conditions. • Place EDG #12 in PTL to prevent overheating.

Answer: C ECW Pump 1B discharge valve is CLOSED and will not OPEN; Place ECW Pump 1B in PTL to prevent running the pump at shutoff head.; Place EDG #12 in PTL to prevent overheating.

Exam Bank No.: 2189

K/A Catalog Number: 076 A2.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(7)

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of SWS

STP Lesson: LOT 201.13 **Objective Number:** 91201

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

Reference: 0POP05-EO-EO00 Addendum 5 step 7 RNO

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Impact is credible if the stated MCC supplied this valve (powered from MCC E1B2). Action is credible because placing CS Pump 1B in PTL would be the correct action if the discharge valve did not open. Since both CS and RCFCs are designed to provide containment cooling, starting RCFCs to replace a CS pump would be appropriate, however in the given situation all RCFCs would already be running.

B: INCORRECT: Impact is credible because some pumps (i.e. AFW Pumps) start with discharge valves open and they must automatically throttle to prevent pump runout, however CS Pump discharge valves are not normally open while in a standby condition. Action is credible because placing CS Pump 1B in PTL would be the correct action if the pump was running in a runout condition. Since both CS and RCFCs are designed to provide containment cooling, starting RCFCs to replace a CS pump would be appropriate, however in the given situation all RCFCs would already be running.

C: CORRECT: MCC E1B3 provides power exclusively to Train B ECW components. The valve would have been closed when it lost power and would not open on pump start. Pump damage could occur while running with discharge closed (no recirc and no cooling). Stopping the DG is also a required action in this case since it is not being provided any cooling water flow.

D: INCORRECT: Impact is credible because some pumps (i.e. AFW Pumps) start with discharge valves open and they must automatically throttle to prevent pump runout, however ECW Pump discharge valves are not normally open while in a standby condition.

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator has to evaluate the given condition to determine the impact and the procedure steps to mitigate the consequences.

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	<u>WHEN</u> the respective ESF Load Sequencer has completed its automatic sequence <u>OR</u> it is determined that the respective ESF Load Sequencer has failed, <u>THEN</u> PERFORM the following: a. Manually START pump(s). b. Manually OPEN discharge isolation valve(s). c. <u>IF</u> any ECW pump can <u>NOT</u> be started <u>OR</u> its discharge isolation valve can <u>NOT</u> be opened, <u>THEN</u>: 1) STOP associated STBY DG by PLACING in PULL-TO-STOP. 2) ENSURE associated Essential Chiller(s) TRIP.
___ 8	VERIFY CCW pumps – RUNNING	<u>WHEN</u> the respective ESF Load Sequencer has completed its automatic sequence <u>OR</u> it is determined that the respective ESF Load Sequencer has failed, <u>THEN</u> manually START pump(s).

Exam Bank No.: 2190 **Last used on an NRC exam:** Never

RO Sequence Number: 56

Unit 1 has experienced a Reactor Trip, Safety Injection and LOOP on Class 1E 4.16 KV Bus Train A. Diesel Generator #11 is supplying power to Class 1E 4.16 KV Bus Train A.

Subsequently AFWP #14 tripped on overspeed. Mechanical Maintenance corrected a mechanical issue with the trip linkage and says that AFWP #14 overspeed trip can now be reset.

Which of the following describes and explains a precaution the Operations Crew should take prior to resetting the overspeed trip for AFWP #14?

A. Reset ESF Load Sequencer Train A and SG LO-LO Actuations to ensure AF-MOV-0514, TURB TRIP/THROT remains closed during and after the reset.

B. Reset Safety Injection and SG LO-LO Actuations to ensure AF-MOV-0514, TURB TRIP/THROT remains closed during and after the reset.

C. Reset Safety Injection and SG LO-LO Actuations to ensure MS-MOV-0143, MN STM ISOL remains closed during and after the reset.

D. Reset ESF Load Sequencer Train A and SG LO-LO Actuations to ensure MS-MOV-0143, MN STM ISOL remains closed during and after the reset.

Answer: B Reset Safety Injection Actuations and SG LO-LO Actuations to ensure AF-MOV-0514, TURB TRIP/THROT remains closed during and after the reset.

Exam Bank No.: 2190

K/A Catalog Number: 013 G2.1.32 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.8 **10CFR Reference:** 55.41(b)(10)

Engineered Safety Features Actuation System: Ability to explain and apply system limits and precautions.

STP Lesson: LOT 202.28 **Objective Number:** 43847

DISCUSS the following elements associated with the AFW turbine driven pump: B. How to reset the trip and throttle valve.

Reference: LOT 202.28 and 0POP02-AF-0002, Resetting Auxiliary Feedwater Pump 14(24) Mechanical Overspeed Trip Device

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because the ESF Load Sequencers provide a start signal to all the other AFW Pumps, but not AFWP #14.

B: CORRECT: AFWP #14 gets a start signal from SI and LO-LO SG level. These two actuations (specifically Train A) must be reset prior to resetting the overspeed trip device to ensure the trip throttle valve, MOV-0514, does not open while resetting.

C: INCORRECT: Credible because MS-MOV-0143, MS Isolation to AFWP #14 does get an open signal upon an AFW actuation, but this valve would already be open (and is normally open).

D: INCORRECT: Reset is credible because the ESF Load Sequencers provide a start signal to all the other AFW Pumps, but not AFWP #14. Explanation is credible because MS-MOV-0143, MS Isolation to AFWP #14 does get an open signal upon an AFW actuation, but this valve would already be open (and is normally open).

Question Level: H **Question Difficulty** 3

Justification:

The Operator must evaluate the given condition and then apply knowledge of the precautions associated with AFWP #14.

4.0 Notes and Precautions

4.1 WHEN stopping turbine driven AFW Pump 14(24), THEN AFW Pump 14(24) SHOULD be tripped to avoid any possibility of operation less than 1900 rpm. (Reference 2.5)

4.2 Components on and around AFW Pump 14(24) MAY be HOT. Caution SHOULD be used when working on or around AFW Pump 14(24).

4.3 Resetting TRAIN A SAFETY INJECTION and TRAIN A SG LO-LO LEVEL actuation signals prior to resetting the overspeed trip device should keep MOV-0514 from automatically opening as soon as the operator resets the overspeed trip device.

5.0 Resetting Auxiliary Feedwater Pump 14(24) Mechanical Overspeed Trip Device

NOTE

All component locations are IVC 10' AFW Pump 14(24) Pump Cubicle unless otherwise noted.

5.1 VERIFY the AFW Pump 14(24) "TURB TRIP/THROT MOV-0514" is CLOSED (CP 006). _____

5.2 VERIFY the AFW Pump 14(24) MOV-0514 Turbine Mechanical Overspeed Trip Linkage is "TRIPPED" by observing the Latch Up Lever **NOT** engaged with the Latch Trip Hook.
(See Addendum 1, Picture 1 for component location and identification) _____

5.3 ENSURE "1(2)-AF-ZSC-7537A, TERRY TURBINE MECHANICAL OVERSPEED TRIP SWITCH" Limit Switch roller is on the MOV-0514 side of the Head Lever (Above the Turbine Pump Casing).
(See Addendum 1 for component location and identification) _____

NOTE

Spring tension pulls the linkage away from the Trip and Throttle Valve.

5.4 RESET the Mechanical Overspeed Trip Linkage by PUSHING/PULLING the Linkage Connecting Rod towards "1(2)-MS-0514 MAIN STM TO TERRY TURBINE THROTTLE MOV". _____

5.5 ENSURE that the Trip Hook and Latch Up Lever are fully engaged.
(See Addendum 1, Pictures 2 and 3 for a proper comparison) _____

Exam Bank No.: 2191 **Last used on an NRC exam:** Never

RO Sequence Number: 57

Unit 1 Control Room is notified by workers excavating in the yard east of the unit that a buried pipe has been damaged.

- A Control Room operator observes IA press PI- 8563 lowering below 95 psig.
- Annunciator IAS HDR PRESS LO 8M03-D3 is received.
- The IA header press stops lowering and begins to rise.

Which of the following correctly identifies what has occurred?

At 90 psig…

A. IA Yard Isolation Valve 1-IA-PV-8568 auto closed to isolate an Instrument Air pipe leak.

B. IA Dryer Emergency Bypass Valve opened to provide sufficient air volume to overcome the leak.

C. SA Isolation Valve 1-IA-PV-9785 auto closed to isolate a Service Air pipe leak.

D. #14 IA Compressor auto started and is supplying sufficient volume to overcome the leak.

Answer: A IA Yard Isolation Valve 1-IA-PV-8568 auto closed to isolate an Instrument Air pipe leak.

Exam Bank No.: 2191

K/A Catalog Number: 078 A3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.1 **10CFR Reference:** 55.41(b)(4)

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

STP Lesson: LOT 202.26 **Objective Number:** 92995

Given a scenario in which Instrument Air pressure is decreasing, PREDICT Instrument and Service Air system component automatic actions that will occur as pressure decreases.

Reference: 0POP04-IA-0001, Loss of Instrument Air

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: 90 psig is the set point for IA HDR PRESS LO alarm and closes IA yard Isol valve

B: INCORRECT: Credible because the IA Emergency Bypass receives an open signal on low pressure (80 psig). IA pressure has not reached the 80 psig setpoint yet.

C: INCORRECT: Credible because the Service Air Isolation Valve receives a closed signal on low pressure (100 psig). If this would have isolated the leak then IA pressure would have started to go back up before it reached 95 psig.

D: INCORRECT: Credible because the 14 IA Compressor receives an auto start signal on low pressure (113 psig). If this would have helped the leak then IA pressure would have started to go back up before it reached 95 psig.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must correlate reported field information and check IA header, recall alarm setpoint and which component will actuate to isolate leak.

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.: 2192

Last used on an NRC exam: Never

RO Sequence Number: 58

Given the Following:

- Unit 2 Control Room is performing 0POP05-EO-EC12 LOCA OUTSIDE CONTAINMENT.
- FHB -4 ft el Area Radiation Monitor is in High Alarm on RM-11
- 'A' Train SI/CS Hi/Hi sump alarm on QDPS

Which of the following describes the impact of this condition and the actions the crew should take to mitigate the consequences?

	IMPACT	ACTIONS
A.	RWST inventory will not be available for recirculation phase.	Place 'A' Train SI/CS pumps in PTL.
B.	Sump tanks in the FHB will have to be transferred to the RCB for recirculation phase.	Align the 'A' Train SI/CS sump to discharge to the Containment Emergency Sump.
C.	RWST inventory will not be available for recirculation phase.	Align the 'A' Train SI/CS sump to discharge to the Containment Emergency Sump.
D.	Sump tanks in the FHB will have to be transferred to the RCB for recirculation phase.	Place 'A' Train SI/CS pumps in PTL.

Answer: A RWST inventory will not be available for recirculation phase. Place 'A' Train SI/CS pumps in PTL.

Exam Bank No.: 2192

K/A Catalog Number: 006 A2.11 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.0 **10CFR Reference:** 55.41(b)(10)

Emergency Core Cooling System:
Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture of ECCS Header

STP Lesson: LOT 201.10 **Objective Number:** 17259

DESCRIBE the flowpath for the ECCS to include major components and valves.

Reference: LOT 201.10 Lesson on ECCS and LOT 504.46 Lesson on 0POP05-EO-EC12 LOCA OUTSIDE CONTAINMENT step 3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: RWST inventory leaked into FHB would not be available for injection into RCS. The FHB radiation monitor alarm and SI/CS pump room sump are indications used to identify a potential leak and location, then the train is secured to try and isolate the leak.

B: INCORRECT: Credible because both given options would allow recovery of lost inventory, however plant design does not support.

C: INCORRECT: The action is credible because it provides a possible method of recovering lost inventory but is not supported by plant design.

D: INCORRECT: Impact is credible because it provides a reasonable action that could recover lost RWST inventory, but is not supported by plant design.

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator must be able to predict the impact of the malfunction and then identify the correct mitigation based on the given information.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

Foldout CIP page should be open.

___1 **CHECK For LOCA In FHB:**

	___a. FHB SI/CS pump sump level alarms - CLEAR	a. GO TO Step 1.c.
	___b. FHB radiation monitors - IN ALARM OR RISING o Area radiation monitors OR o Ventilation radiation monitors	b. GO TO Step 5.
	___c. RCS hot leg sample ICIVs - CLOSED: o "RCS LOOP 1A(2A)" "Th SAMPLE ICIV FV-4454" o "RCS LOOP 1C(2C)" "Th SAMPLE ICIV FV-4455"	c. CLOSE RCS sample isolations.
	___d. RHR sample ICIV - CLOSED: o "RHR SAMPLE" "ICIV FV-4823"	d. CLOSE RHR sample isolations.

___2 **RESET SI AUTO RECIRC**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___3	**ISOLATE SI Train With High Sump Level:**	
	___a. CHECK FHB ECCS sump levels - ONLY ONE TRAIN IN HI-HI ALARM	a. GO TO Step 4.

	___b. PLACE pumps in affected train in PULL TO LOCK o RHR pump o LHSI pump o HHSI pump o CS pump	
	___c. CHECK affected LHSI pump cold leg injection valve - CLOSED: o 1(2)-RH-MOV-0031A o 1(2)-RH-MOV-0031B o 1(2)-RH-MOV-0031C	c. PERFORM the following: 1) ENERGIZE affected LHSI pump cold leg injection valve. a) DISPATCH Operator To UNLOCK and CLOSE the selected LHSI Cold Leg Injection Valve Breakers: o "LHSI PUMP 1A(2A) DISCH" "TO LOOP 1 COLD LEG" "1(2)-SI-MOV-0031A" E1A1(E2A1) / C3 & V6L o "LHSI PUMP 1B(2B) DISCH" "TO LOOP 2 COLD LEG" "1(2)-SI-MOV-0031B" E1B1(E2B1) / A4 & W2R o "LHSI PUMP 1C(2C) DISCH" "TO LOOP 3 COLD LEG" "1(2)-SI-MOV-0031C" E1C1(E2C1) / D1 & S4R 2) CLOSE affected LHSI pump cold leg injection valve.

Step 3 continued on next page.

Exam Bank No.: 2193 **Last used on an NRC exam:** Never

RO Sequence Number: 59

Unit 1 has experienced a Reactor Trip from 100% power. The AFW System has actuated but no Operator actions have been performed yet.

Which of the following describes the effect on the AFW System if a QDPS #2 APC were to lose power?

A. All AFW Pump Flow Regulating Valves would fail in the AS IS POSITION.

B. Only the AFW Pump Flow Regulating Valve associated with the failed #2 APC would fail in the AS IS POSITION.

C. All AFW Pump Outside Containment Isolation Valves would fail in the AS IS POSITION.

D. Only the AFW Pump Outside Containment Isolation Valve associated with the failed #2 APC would fail in the AS IS POSITION.

Answer: B Only the AFW Pump Flow Regulating Valve associated with the failed #2 APC would fail in the AS IS POSITION.

Exam Bank No.: 2193

K/A Catalog Number: 016 K3.06 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(7)

Knowledge of the effect that a loss or malfunction of the NNIS will have on the following: AFW System

STP Lesson: LOT 202.44 **Objective Number:** 7667

Given a change in plant or system condition EXPLAIN the operation and indications of the QDPS System.

Reference: LOT 202.44 QDPS and LOT 202.28 AFW System

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because each reg valve receives a control signal, but not all from the same QDPS APC.

B: CORRECT: QDPS #2 APCs control their associated train of AFW Reg Valve.

C: INCORRECT: Credible because the AFW Outside Containment Isolation Valves automatically receive an open signal upon an actuation, but it comes from a source other than the QDPS APC (SSPS).

D: INCORRECT: Credible because the AFW Outside Containment Isolation Valves automatically receive an open signal upon an actuation, but it comes from a source other than the QDPS APC (SSPS) and the APCs are train related.

Question Level: F **Question Difficulty** 3

Justification:

The Reactor Operator must have fundemental knowledge of two systems; QDPS and AFW.

The Upper Section of the # 2 APCs provides the Valve Control

- Automatic and Manual control of the SG PORVs.
- Automatic control of the AFW throttle valves.
- Manual control of the Reactor Head Vent Valves.

Exam Bank No.: 2194 **Last used on an NRC exam:** Never

RO Sequence Number: 60

Following a Loss of Coolant Accident (LOCA), the pH of the fluid used by the Containment Spray pumps is controlled to reduce corrosion and maintain iodine in solution.

Which of the following describes how the pH of the Containment Spray fluid is controlled?

A. Minimum boric acid concentration requirements for the Refueling Water Storage Tank (RWST) ensure the proper pH is maintained in the emergency sumps.

B. Sodium Hydroxide (NaOH) from the Spray Additive Tanks mixes with the Containment Spray Pump discharge.

C. Powdered trisodium phosphate stored in six baskets located on the -11 foot elevation of the RCB dissolves into the fluid which flows into the emergency sumps.

D. Lithium Hydroxide (LiOH) is injected into the emergency sumps.

Answer: C Powdered trisodium phosphate stored in six baskets located on the -11 foot elevation of the RCB dissolves into the fluid which flows into the emergency sumps.

Exam Bank No.: 2194

K/A Catalog Number: 026 K4.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.1 **10CFR Reference:** 55.41(b)(7)

Knowledge of CSS design feature(s) and/or interlock(s)
which provide for the following: Neutralized boric acid to reduce corrosion and remove inorganic fission product iodine from steam (NAOH) in containment spray

STP Lesson: LOT 201.11 **Objective Number:** 29767

STATE the name and the function of the chemical used in the Recirculation Fluid pH Control System

Reference: LOT201.11 handout page 7 & 8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - Credible because boric acid will change pH (lower it), however the additive used raises pH.

B: INCORRECT - Credible because spray additive tanks were part of original plant design (and are still physically in the plant), but system was abandoned in place and replaced by the trisodium phosphate baskets.

C: CORRECT - baskets of trisodium phosphate dissolve into the fluid on the containment floor to raise the pH of the fluid.

D: INCORRECT - Credible because LiOH is used during normal operation to control the pH of the RCS.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have knowledge of the design and function of the recirculation fluid pH control system.

Automatic Actions: None

Immediate Actions: None

Probable Causes: Electrical Fault

Breaker opened locally

Setpoint: Breaker open and no bus strip signal and control switch not in STOP or PTL

2. CS Pump Discharge Flow Lo

Automatic Actions: None

Immediate Actions: None

Probable Causes: Improper valve lineup

Loss of suction

Instrument failure

Setpoint: 500 gpm

3. ESF bypass/INOP

a. Pull-To-Lock

b. Breaker not racked in

c. Loss of control power

4. Fail To Actuate - CS Actuation Signal and Pump not running

RECIRCULATION FLUID PH CONTROL SUB-SYSTEM

Six Stainless Steel Baskets (approximately 4' X 8') are filled with powdered Tri-Sodium Phosphate (TSP). When dissolved by the break flow of a LOCA or major Main Steam Line Break (MSLB) in addition to Containment Spray Flow and SI Flow, the PH of the fluid collected in the three (3) Containment Emergency Sumps is increased to 7.0 - 9.5.

The Boric Acid in the RWST and RCS can cause the PH of the Fluid in the Sumps to be as low as 4.5. Tech Specs requires that between 11,500 lbs to 15,100 lbs of TSP be available for mixing with the SI Injection Fluid and Break Fluid. This guarantees that a Fluid PH of 7.0 to 9.5 is achieved in the three (3) Containment Emergency Sumps.

Controlling PH ≥ 7.0 ensures that entrapped iodine remains in solution. Maintaining PH ≤ 9.5 minimizes Chloride Induced Stress Corrosion Cracking of Austenitic Stainless Steel.

The six (6) TSP baskets are located at the following azimuths on the 11' elevation in the RCB:

Inside the Biological Shield - at 45°, 90° and 330°

Outside the Biological Shield - at 140°, 227° and 270°

(NOTE: 0° is at Plant East.)

To ensure that each basket is filled to the proper level, an "Indicator" mark is checked during each refueling outage. This satisfies the Tech. Spec. Surveillance.

SPRAY HEADERS & NOZZLES

The CS Pump Discharge Headers penetrate Containment and join a common Spray Header climbing up the Containment Interior Wall to the Spray Rings. The Common Spray Header supplies 4 concentric rings and associated nozzles. The nozzles are drilled with 3/8” diameter orifices allowing 1/3” particle passage. The nozzle locations and orientation angles ensure a minimum of 90 percent Containment coverage. The Spray Ring is located as high as possible in the RCB without incurring Spray Pattern Interference.

Ring 1 is a 4” diameter ring with a 13’ radius, 225’ elevation and 21 nozzles.

Ring 2 is a 6” diameter ring with a 26’ radius, 221’ elevation and 49 nozzles.

Ring 3 is a 6” diameter ring with a 45’ radius, 210’ elevation and 59 nozzles.

Ring 4 is an 8” diameter ring with a 64’ radius, 188’ elevation and 119 nozzles.

Exam Bank No.: 2195 **Last used on an NRC exam:** Never

RO Sequence Number: 61

During the response to a Unit 2 Reactor Trip, all Auxiliary Feedwater was lost. The crew is performing 0POP05-EO-FRH1, Response to Loss of Secondary Heat Sink, Addendum #1, Establishing Main Feedwater Flow.

The following conditions exist:

- RCS Feed and Bleed has been established.
- RCS Wide Range T_H is 600ºF and rising.
- All SG Wide Range levels are 11% and lowering.
- The crew is ready to feed Main Feedwater to the Steam Generators.

Which of the following describes the IMPACT of feeding Main Feedwater to dry Steam Generators and the REQUIRED ACTION from 0POP05-EO-FRH1, Response to Loss of Secondary Heat Sink, which should minimize the IMPACT?

	IMPACT	REQUIRED ACTION
A	Steam Generator integrity will be impacted due to thermal shock.	If CETs are rising establish maximum flow rate to ALL SGs.
B	Steam Generator integrity will be impacted due to high pressure.	If CETs are rising establish maximum flow rate to ONLY ONE SG.
C	Steam Generator integrity will be impacted due to thermal shock.	If CETs are rising establish maximum flow rate to ONLY ONE SG.
D	Steam Generator integrity will be impacted due to high pressure.	If CETs are rising establish maximum flow rate to ALL SGs.

Answer: C Steam Generator integrity will be impacted due to thermal shock. If CETs are rising establish maximum flow rate to ONLY ONE SG.

Exam Bank No.: 2195

K/A Catalog Number: 059 A2.04 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(5)

Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feeding a Dry Steam Generator

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 - Lesson on 0POP05-EO-FRH1, Response to Loss of Secondary Heat Sink. See CIP

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because the given RCS temperature is high for post trip conditions (and rising), so the first instinct is to lower temperature as quickly as possible.

B: INCORRECT: Credible because adding water to a hot vessel will quickly produce steam and raise pressure, however the SG safeties are still available and designed to protect the SG from a high pressure, loss of integrity event.

C: CORRECT: The impact of feeding a dry SG is thermal shock. Because of this feedwater is established to only one SG at a time. The heat removal capability from one SG is greater than decay heat.

D: INCORRECT: Impact is credible because adding water to a hot vessel will quickly produce steam and raise pressure, however the SG safeties are still available and designed to protect the SG from a high pressure, loss of integrity event. Action is credible because the given RCS temperature is high for post trip conditions (and rising), so the first instinct is to lower temperature as quickly as possible.

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator needs to have fundamental knowledge of the basis for cautions in emergency procedures and, when given information, determine the required actions from the procedure to take.

CONDITIONAL INFORMATION PAGE

RCS BLEED AND FEED CRITERIA AFTER STEP 1
WITH a heat sink required per STEP 1, IF SG wide range levels on any TWO SGs are LESS THAN 50% [73%] OR pressurizer pressure is GREATER THAN OR EQUAL TO 2335 psig due to loss of secondary heat sink, THEN TRIP RCPs and GO TO Step 10 to initiate bleed and feed to prevent core damage.

FEED AN INTACT SG FIRST
IF an intact SG is available, THEN DO NOT re-establish feed flow to any faulted SG to prevent excessive RCS cooldown.

MANUAL SI ACTUATION
IF plant conditions degrade, THEN manual SI actuation is required following block of automatic SI actuation.

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.

CONTAINMENT SPRAY ACTUATION CRITERIA
IF containment pressure rises to GREATER THAN 9.5 PSIG, THEN VERIFY the following:
- o Containment Spray Actuation
- o Phase B Isolation

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.

CONTAINMENT SPRAY / PHASE B ACTUATION
IF a Containment Spray / Phase B actuation occurs, THEN STOP all RCPs.

SG FEED FLOW CRITERIA
IF SG WR level is GREATER THAN 13%[37%] and a feedwater source becomes available, THEN RESTORE feed flow to **ONLY** SG(s) with WR level GREATER THAN 13%[37%].

IF ALL SG's WR levels LESS THAN 13%[37%] AND RCS WR Thot LESS THAN 550°F AND Bleed/Feed is in progress, THEN when feedwater is available, ESTABLISH flow rate to **ONLY** one SG NOT to exceed 100 gpm, WHEN SG WR level GREATER THAN 13%[37%], THEN ADJUST flow rate as needed to restore NR level.

IF ALL SG's WR levels LESS THAN 13%[37%], RCS WR Thot GREATER THAN 550°F and Bleed/Feed is in progress, THEN when feedwater is available, RESTORE flow as follows:
- o IF CET temperatures are rising, THEN ESTABLISH maximum flow rate to **ONLY** one SG.
- o IF CET temperatures are stable or lowering, THEN ESTABLISH flow rate to **ONLY** one SG NOT to exceed 100 gpm. WHEN RCS WR Thot LESS THAN 550°F AND SG WR level GREATER THAN 13%[37%], THEN ADJUST flow rate as needed to restore NR level.

Exam Bank No.: 2196 **Last used on an NRC exam:** Never

RO Sequence Number: 62

Given the following:

- A Unit 1 reactor trip has occurred.
- An AFW actuation occurred as expected.
- The pump coupling on AFW Pump #13 broke causing the pump shaft to separate from the motor shaft.

Based on these conditions, which of the following is true?

A. AFWP 13 TRIP alarm will annunciate due to motor overspeed after the coupling breaks.

B. AFWP 13 TRIP alarm will annunciate when pump flow drops to <90 gpm after the coupling breaks.

C. AFWP 13 DISCH PRESS LO alarm will annunciate, the motor should be stopped by placing the hand switch in the STOP position.

D. AFWP 13 DISCH PRESS LO alarm will annunciate, the motor should be stopped by placing the hand switch in the PTL position.

Answer: D AFWP 13 DISCH PRESS LO alarm will annunciate, the motor should be stopped by placing the hand switch in the PTL position.

Exam Bank No.: 2196

K/A Catalog Number: 061 K6.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(7)

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Pumps

STP Lesson: LOT 202.28 **Objective Number:** 43805

DESCRIBE the AFW system controls and instrumentation in the MCR.

Reference: POP09-AN-06M4 pages 15 and 28 & Logic 9Z40131

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - Credible because turbine driven and engine driven pumps have overspeed trips and motor speed will temporarily rise when the coupling breaks.

B: INCORRECT - Credible because 90 gpm is the point where the recirc valve begins to open and low flow is often indicative of the need for a pump trip.

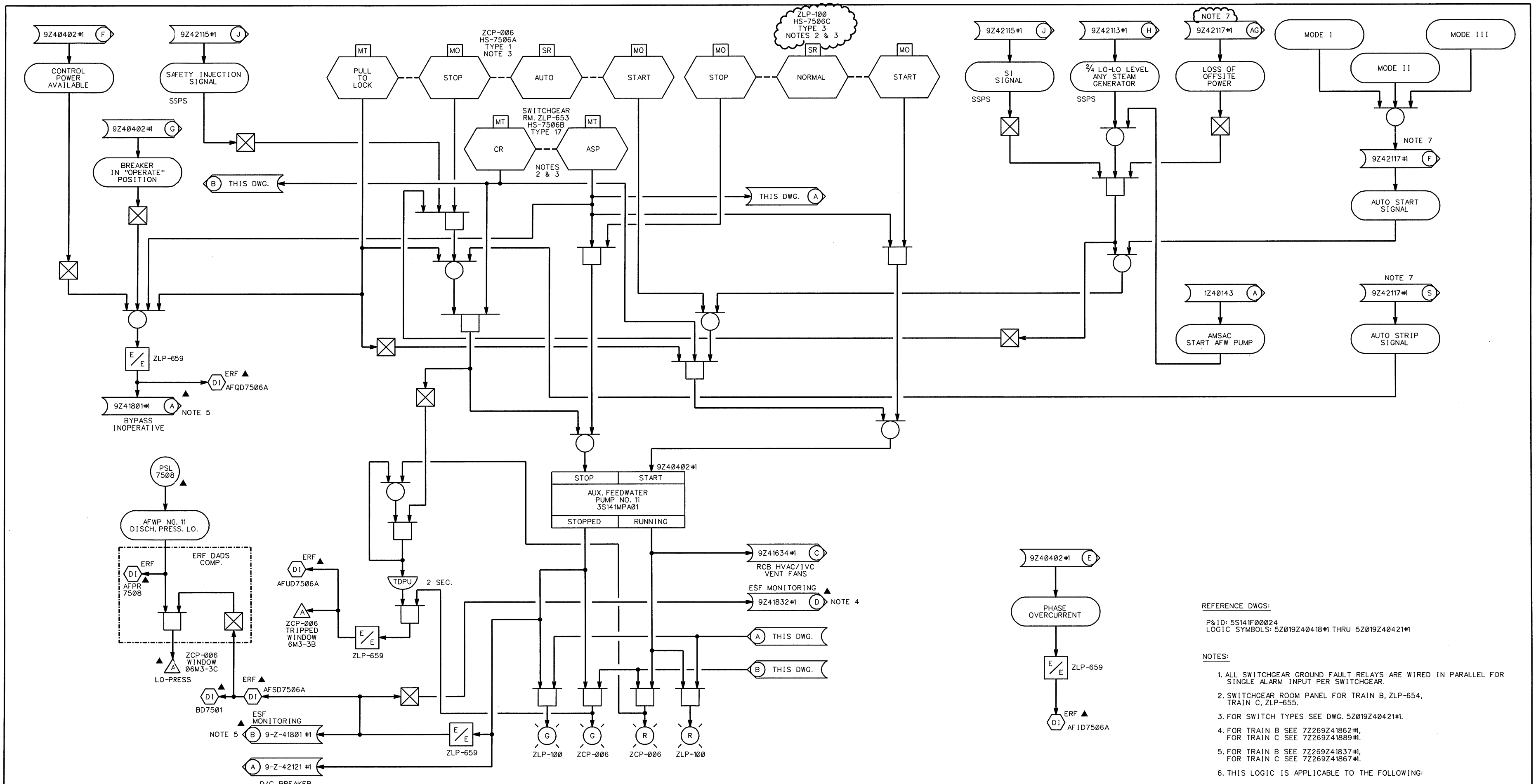
C: INCORRECT - Credible because normally, placing a switch in "stop" will secure the pump, but since an AFW actuation is present (given) the switch must be taken to PTL.

D: CORRECT - A low discharge pressure condition will occur when the coupling breaks and the pump shaft stops. The motor can only be stopped in this condition by placing the handswitch in PTL due to the AFW actuation signal present.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have a fundamental knowledge of the effects of a shaft shear on pump operation and how it relates to the indications available for the AFW pumps.



REFERENCE DWGS:

P&ID: 5S141F00024
LOGIC SYMBOLS: 5Z019Z40418#1 THRU 5Z019Z40421#1

NOTES:

1. ALL SWITCHGEAR GROUND FAULT RELAYS ARE WIRED IN PARALLEL FOR SINGLE ALARM INPUT PER SWITCHGEAR.
2. SWITCHGEAR ROOM PANEL FOR TRAIN B, ZLP-654, TRAIN C, ZLP-655.
3. FOR SWITCH TYPES SEE DWG. 5Z019Z40421#1.
4. FOR TRAIN B SEE 7Z269Z41862#1, FOR TRAIN C SEE 7Z269Z41889#1.
5. FOR TRAIN B SEE 7Z269Z41837#1, FOR TRAIN C SEE 7Z269Z41867#1.
6. THIS LOGIC IS APPLICABLE TO THE FOLLOWING:

COMPUTER POINT							PLANT COMP.	SERVICE	SEQUENCE ACTUATION	ESF TRAIN	ESF BYPASS STATUS INOP.	CONTROL SWITCH	TRANSFER SWITCH	CONTROL SWITCH	ANNUNCIATOR LOCATION	ISOLATION CABINET	ANNUN. TRIPPED	ELECT. REF. DWG. NO.	MECHANICAL EQUIP. NO.
BYP./INOP.	OVERCURRENT	DISCH. PRESS LO	ANN. LO PRESS.	STOPPED	TRIPPED	PRESS SW.	TRIPPED												
AFQD7506A	AFID7506A	AFPR7508	06M3-3C	AFSD7506A	AFUD7506A	PSL-7508	BD7501	AFW PUMP 11	5Z019Z42117#1	A	9-Z-41801 #1	HS-7506A ZCP-006	HS-7506B ZLP-653	HS-7506C ZLP-100	ZCP-006	ZLP-659 (SEE NOTE 8)	06M3-3B	00009E0AF01#1 SHT. 01	3S141MPA01
AFQD7507A	AFID7507A	AFPR7507	06M3-7C	AFSD7507A	AFUD7507A	PSL-7507	BD7503	AFW PUMP 12	5Z019Z42118#1	B	9-Z-41837 #1	HS-7507A ZCP-006	HS-7507B ZLP-654	HS-7507C ZLP-100	ZCP-006	ZLP-660 (SEE NOTE 8)	06M3-7B	00009E0AF01#1 SHT. 01	3S141MPA02
AFQD7508A	AFID7508A	AFPR7506	06M4-3C	AFSD7508A	AFUD7508A	PSL-7506	BD7505	AFW PUMP 13	5Z019Z42119#1	C	9-Z-41867 #1	HS-7508A ZCP-006	HS-7508B ZLP-655	HS-7508C ZLP-100	ZCP-006	ZLP-661 (SEE NOTE 8)	06M4-3B	00009E0AF01#1 SHT. 01	3S141MPA03

7. SEE DWGS. 5Z019Z42118#1 FOR TRAIN B & 5Z019Z42119#1 FOR TRAIN C.
8. THIS CABINET CONSISTS OF 3 BAYS: A, B AND C.

GENERAL NOTES:

1. THIS DRAWING REPRESENTS FUNCTIONAL REQUIREMENTS ONLY.

NUCLEAR SAFETY RELATED | Q

NO.	ISSUE DATE	REVISION	ORIG	CKR	RE	DV	SE
14	5-24-12	REVISED PER CR 12-16733-7.	VJM	[illegible]	N/A	N/A	N/A
13	10-19-11	REVISED PER DCP 07-11358-1.	VJM	ON FILE			
12	12-11-96	ENHANCEMENT & EDITORIAL	ELT	ON FILE			
0-11	6-23-83	ISSUED FOR CONSTRUCTION	SIGNATURES ON FILE				

SOUTH TEXAS PROJECT
NUCLEAR OPERATING COMPANY
STP

MOTOR DRIVEN AFW PUMP
LOGIC DIAGRAM
SYSTEM: AF

PRIORITY	CADD FILE NO.	FSUG. NO.	STI. NO.	SCALE	DWG. NO.	SHT.	REV.
1	Z1048.141	D070928	33554872	NONE	5S149Z40131#1		14

DSEED.DGN

5/23/2012 10:53:31 AM

Exam Bank No.: 2197 **Last used on an NRC exam:** Never

RO Sequence Number: 63

A lockout has occurred on 4.16 KV bus E2A. The Unit Supervisor has directed you to determine the E2A11 battery discharge current.

Which of the following correctly describes how you should obtain this information for the Unit Supervisor?

This information can …

A. be obtained from CP-003 using the BATT CUR indicator.

B. be obtained from CP-010 using the BATT CUR indicator.

C. only be obtained from one of the QDPS plasma displays.

D. only be obtained locally. A Plant Operator must be dispatched.

Answer: A be obtained from CP-003 using the BATT CUR indicator.

Exam Bank No.: 2197

K/A Catalog Number: 063 A4.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.0 **10CFR Reference:** 55.41(b)(7)

Ability to manually operate and/or monitor in the control room: Battery discharge rate

STP Lesson: LOT 201.37 **Objective Number:** 92986

DESCRIBE the local and MCR instrumentation available to monitor the Class 1E 125 VDC System

Reference: LOT201.37 PowerPoint slide 55

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT - This information is available on CP-003

B: INCORRECT - Credible because CP-010 is also an electrical panel, however this indication is not located on it.

C: INCORRECT - Credible because the QDPS computer provides safety related system information, but not this. This information is available on the ICS computer system (in the control room).

D: INCORRECT - Credible because many plant parameters are only available through local indication. Tha applicant must be familiar with what indication is on the control panels to correctly respond.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have knowledge of the indications available in the control room for the batteries.

Control Room Alarms/Indications

Exam Bank No.: 2198 **Last used on an NRC exam:** Never

RO Sequence Number: 64

Given the following:

- Unit 1 is in Mode 3.
- All Shutdown Rods are fully withdrawn preparing for a Reactor Startup.
- Pressurizer Backup Heaters 'D' and 'E' are energized.

Subsequently:

- "D" RCP Trips

Which of the following describes the correct INITIAL response of Pressurizer Pressure Control and Pressurizer Temperature?

	Pressurizer Pressure Master Controller Output	Pressurizer Temperature
A.	Rise	Rise
B.	Rise	Lower
C.	Lower	Lower
D.	Lower	Rise

Answer: A Rise, Rise

Exam Bank No.: 2198

K/A Catalog Number: 004 A3.15 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the CVCS, including:
PZR pressure and temperature

STP Lesson: LOT 201.14 **Objective Number:** 92779

GIVEN plant conditions, DETERMINE their effects on the Pressurizer pressure and level control system.

Reference: LOT 201.02, rev 10 & LOT 201.14 rev 14

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: PZR spray line connects to the 'D' RCS loop. With the 'D' RCP tripped, PZR spray flow wil lower causing PZR pressure to rise. Master Controller output will rise which will open spray valves to replace lost spray flow. PZR temperature will rise with pressurizer pressure.

B: INCORRECT: This distractor is credible because with a RCP trip, there is a loss of heat input which could be thought to lower PZR pressure and temperature and it shows lack of knowledge of the PZR Pressure Control system response to a drop in pressure. If PZR pressure and temperature go down then controller output will lower NOT rise. The loss of spray flow will have more affect on the PZR Pressure control than the loss of RCP heat input.

C: INCORRECT: This distrator is credible because the master controller output will change and the applicant must understand system operation to determine how it will change. With a RCP trip there is a loss of heat input which could be thought to lower PZR pressure and temperature due to the outsurge, but the loss of spray flow will have more affect on the PZR Pressure control than the loss of heat input to the RCS.

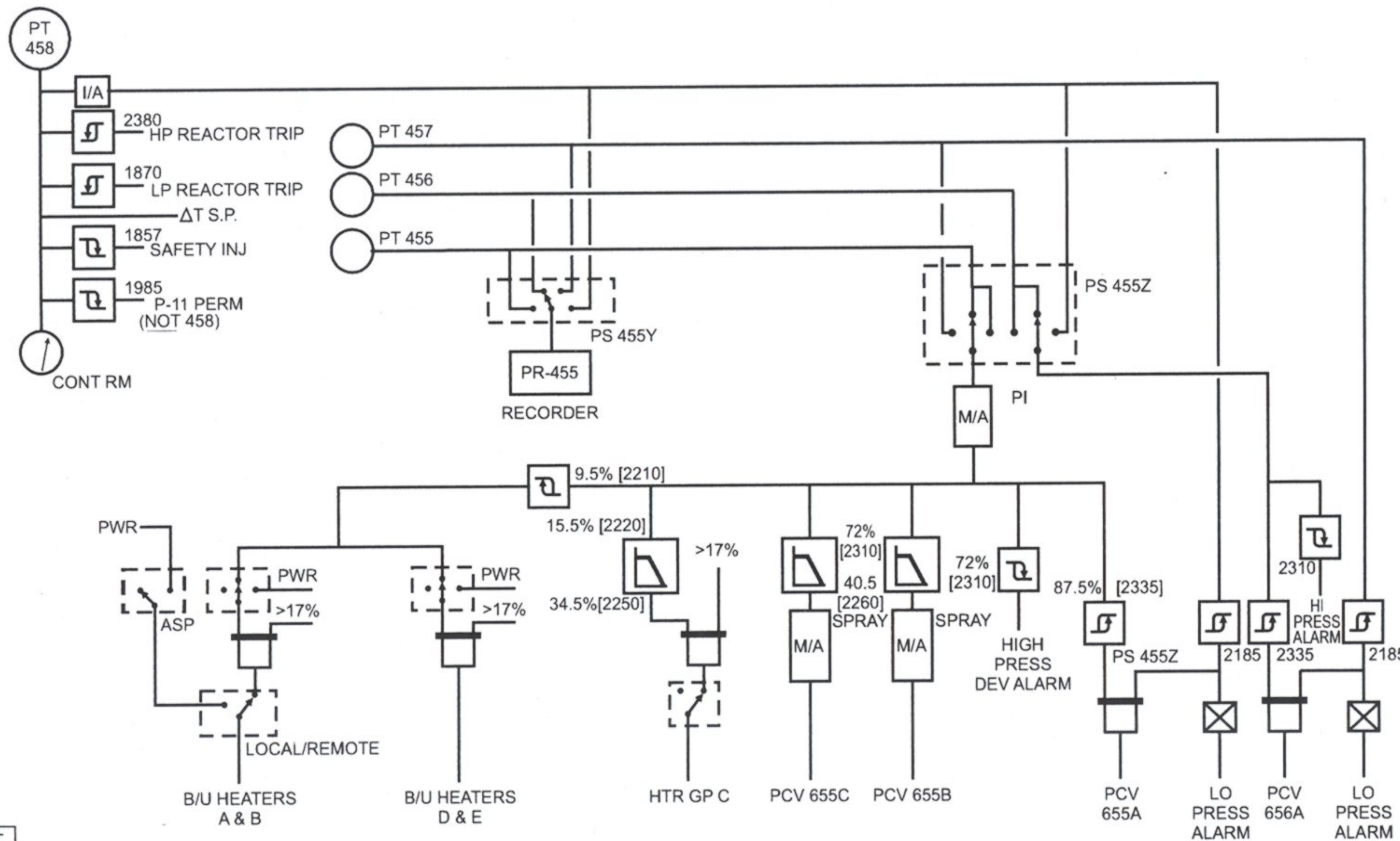
D: INCORRECT: This distrator is credible because the master controller output will change and the applicant must understand system operation to determine how it will change. If PZR pressure and temperature go up then controller output will rise not lower.

Question Level: H **Question Difficulty** 3

Justification:

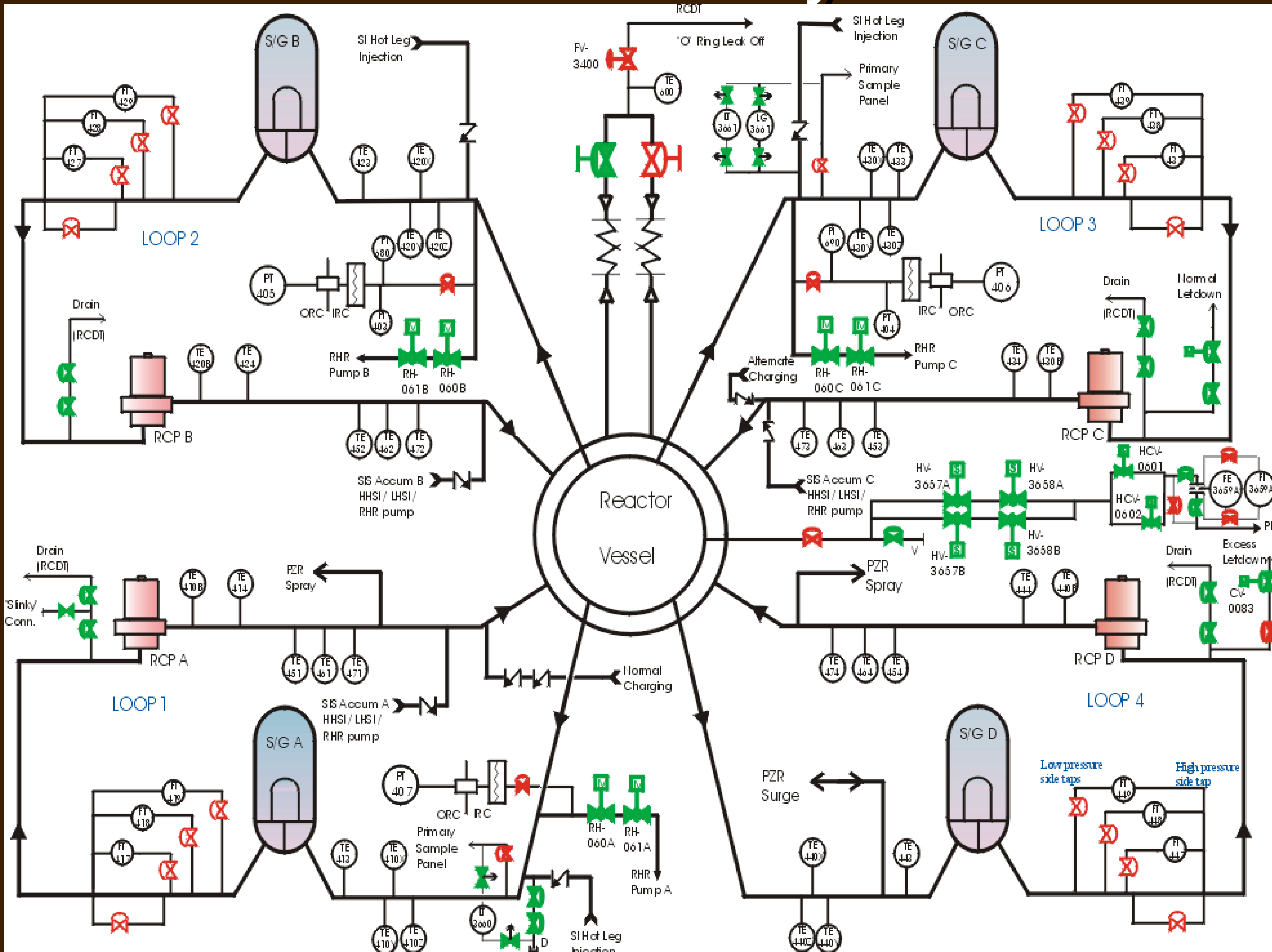
The applicant must analyze the given condition and recall that pressurizer spray is on the D RCS loop and there would be lower flow to the spray nozzles and then analyze the effect on pressurizer pressure control system in auto.

PRESSURIZER PRESSURE CONTROL SYSTEM

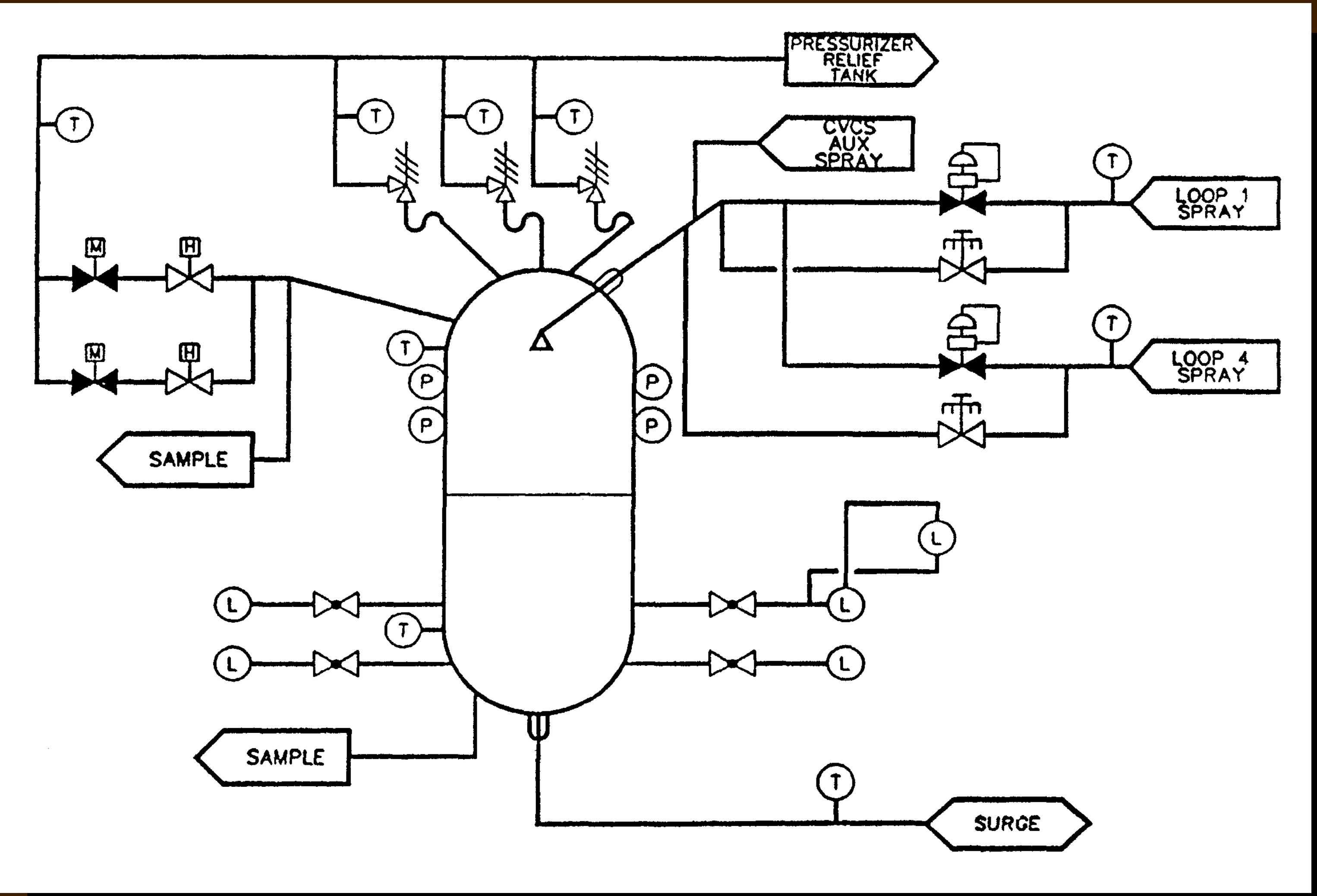


LOT201.14.TP.01
CDR-08/18/99

Reactor Coolant System



Simplified Drawing Prressurizer



Exam Bank No.: 2199 **Last used on an NRC exam:** Never

RO Sequence Number: 65

Unit 1 is at 75% power. The crew is currently raising power at 10%/Hour.

The Primary Reactor Operator over the last 10 minutes has noticed RCP Motor Upper and Lower Radial Bearing Temperatures trending up. The bearing temperatures were all at 140ºF and are now stable at the following temperatures:

	Motor Upper Radial Bearing Temperature	Motor Lower Radial Bearing Temperature
RCP 1A	200ºF	200ºF
RCP 1B	200ºF	190ºF
RCP 1C	190ºF	190ºF
RCP 1D	190ºF	180ºF

At a minimum, which of the following should the Unit Supervisor have the Reactor Operator perform in accordance with 0POP04-RC-0002, Reactor Coolant Pump Off Normal?

Trip the Reactor, Ensure Main Turbine tripped and stop…

A. RCP 1A only

B. RCP 1A and 1B only

C. RCP 1A, 1B and 1C only

D. RCP 1A, 1B, 1C and 1D

Answer: B RCP 1A and 1B only

Exam Bank No.: 2199

K/A Catalog Number: 003 A1.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP pump and motor bearing temperatures.

STP Lesson: LOT 201.05 **Objective Number:** 97119

Given plant conditions, ANALYZE the conditions and accurately PREDICT Reactor Coolant Pump response.

Reference: LOT 201.05 Lesson on RCPs and LOT 501.01 Lesson on 0POP04-RC-0002, Reactor Coolant Pump Off Normal, CIP

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because all given temperatures are above the normal operating range (~140 degrees), so the applicant must have knowledge of the limit to correctly respond.

B: CORRECT: With the given information RCPs 1A and 1B will have to be tripped. If either RCP Motor Upper OR Lower Radial Bearing temperature is above 195 degrees F then the RCP is required to be tripped.

C: INCORRECT: Credible because all given temperatures are above the normal operating range (~140 degrees), so the applicant must have knowledge of the limit to correctly respond.

D: INCORRECT: Credible because all given temperatures are above the normal operating range (~140 degrees), so the applicant must have knowledge of the limit to correctly respond.

Question Level: F **Question Difficulty** 3

Justification:

The Reactor Operator needs to have fundamental knowledge of those parameters that require RCPs to be tripped.

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. CONTINUE at Step 1.0 of procedure

- 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
- 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
- Case Vibration - Mtr_Accel-Vert 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 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

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 this procedure.

RCP TRIP CRITERIA FOR LOSS OF SEAL INJECTION AND LOSS OF THERMAL BARRIER CCW

IF an RCP experiences a simultaneous loss of seal water injection flow AND loss of CCW flow to thermal barrier, THEN STOP affected RCP within 1 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. 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.

Exam Bank No.: 2200 **Last used on an NRC exam:** Never

RO Sequence Number: 66

Unit 2 is performing a Plant Startup with Reactor Power currently at 8%.

Power Range Channel N41 INSTRUMENT and CONTROL power fuses are removed while maintenance is being performed on the detector. All protective bistables associated with N41 are in the TRIPPED condition.

Subsequently the INSTRUMENT power is lost to Power Range Channel N42.

Which of the following describes the IMPACT of this malfunction and the ACTION taken to mitigate the consequences?

	IMPACT	ACTION
A.	All protective bistables associated with N42 will BYPASS.	Manually control LPFRVs to respond to the Steam Generator level transient.
B.	All protective bistables associated with N42 will TRIP.	Perform immediate actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection, to respond to the Reactor Trip.
C.	All protective bistables associated with N42 will BYPASS.	Perform immediate actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection, to respond to the Reactor Trip.
D.	All protective bistables associated with N42 will TRIP.	Manually control LPFRVs to respond to the Steam Generator level transient.

Answer: B All protective bistables associated with N42 will TRIP. Perform the immediate actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection, to respond to the Reactor Trip.

Exam Bank No.: 2200

K/A Catalog Number: 012 A2.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(7)

Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of instrument power.

STP Lesson: LOT 201.20 **Objective Number:** 507227

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

Reference: LOT 201.20 lesson on the Solid State Protection System and LOT 201.16 lesson on Excore NIS

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Impact is credible because the Nis have a bypass feature (however it is manually initiated) and some actuations have been changed from de-energize to actuate to energize to actuate (main steam and feedwater isolations). Action is credible because Power Range NIs do have an input to LPFRV automatic control (and are taken to manual in the off-normal for NIs).

B: CORRECT: The bistables for NIs actuate to the tripped condition when instrument or control power is lost. With N41 bistables in the tripped condition the Reactor will trip.

C: INCORRECT: Impact is credible because the Nis have a bypass feature (however it is manually initiated) and some actuations have been changed from de-energize to actuate to energize to actuate (main steam and feedwater isolations).

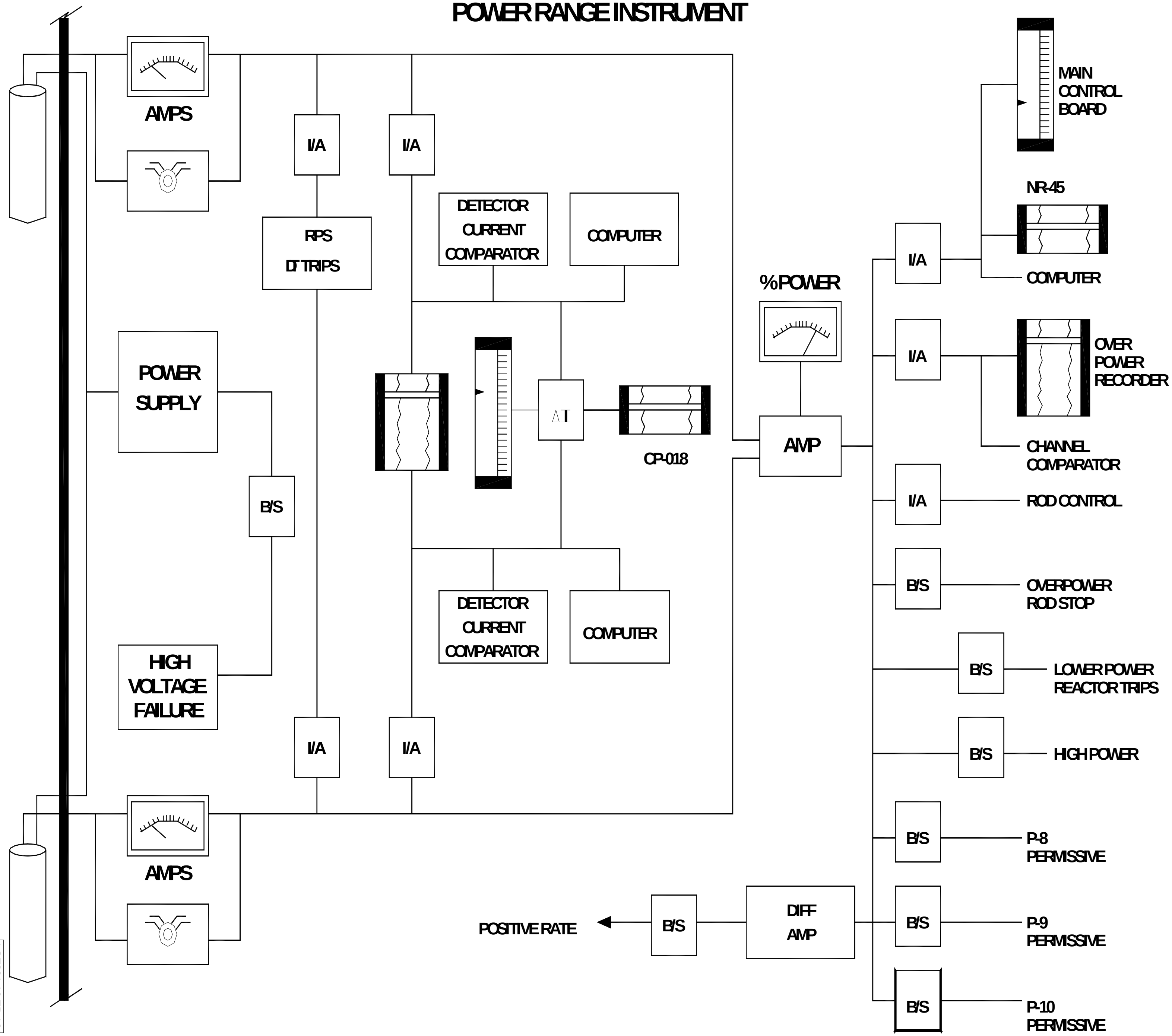
D: INCORRECT: Action is credible because Power Range NIs do have an input to LPFRV automatic control (and are taken to manual in the off-normal for NIs) and several other reactor trips are bypassed when less than 10% (i.e. pressurizer pressure low, pressurizer level high, RCP underfrequency).

Question Level: H **Question Difficulty** 3

Justification:

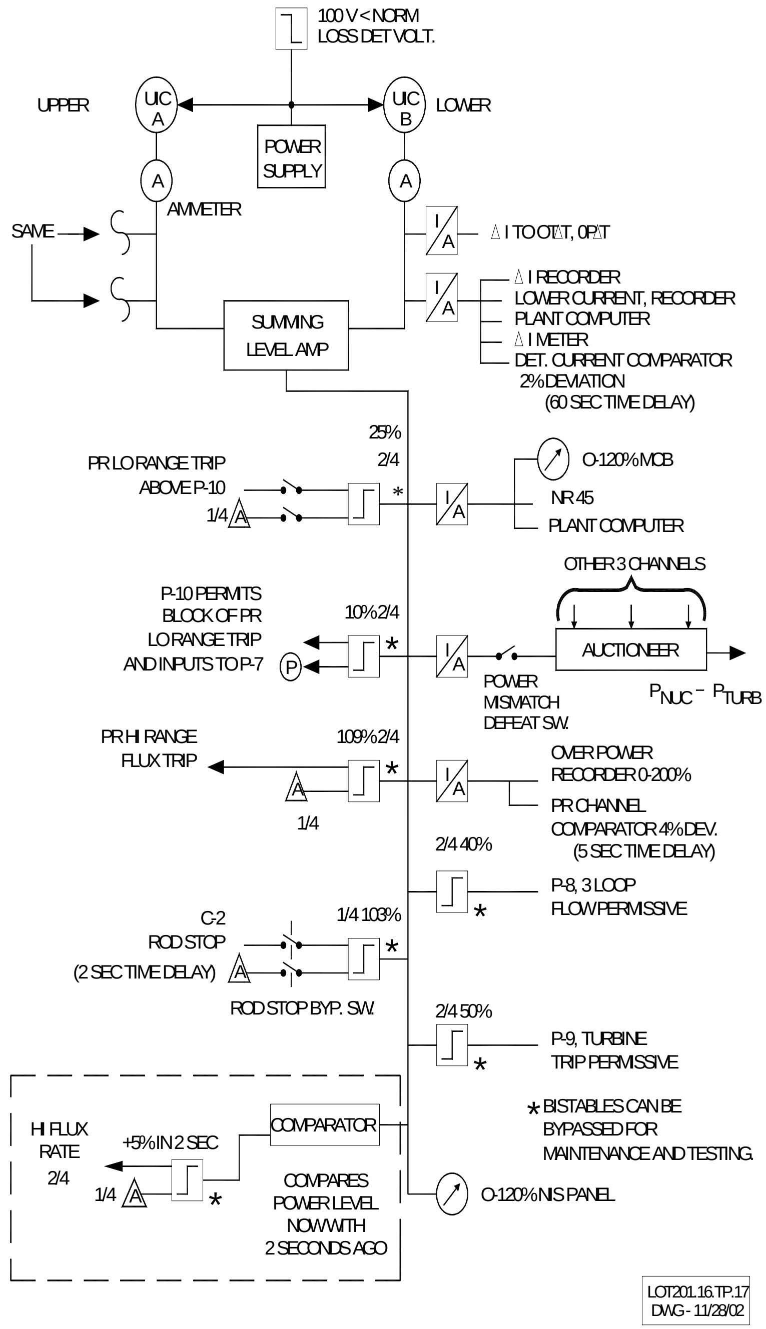
The Reactor Operator must have knowledge of how instrument power for the NIs can affect the SSPS and be able to evaluate the given condition to determine the correct action to implement.

POWER RANGE INSTRUMENT

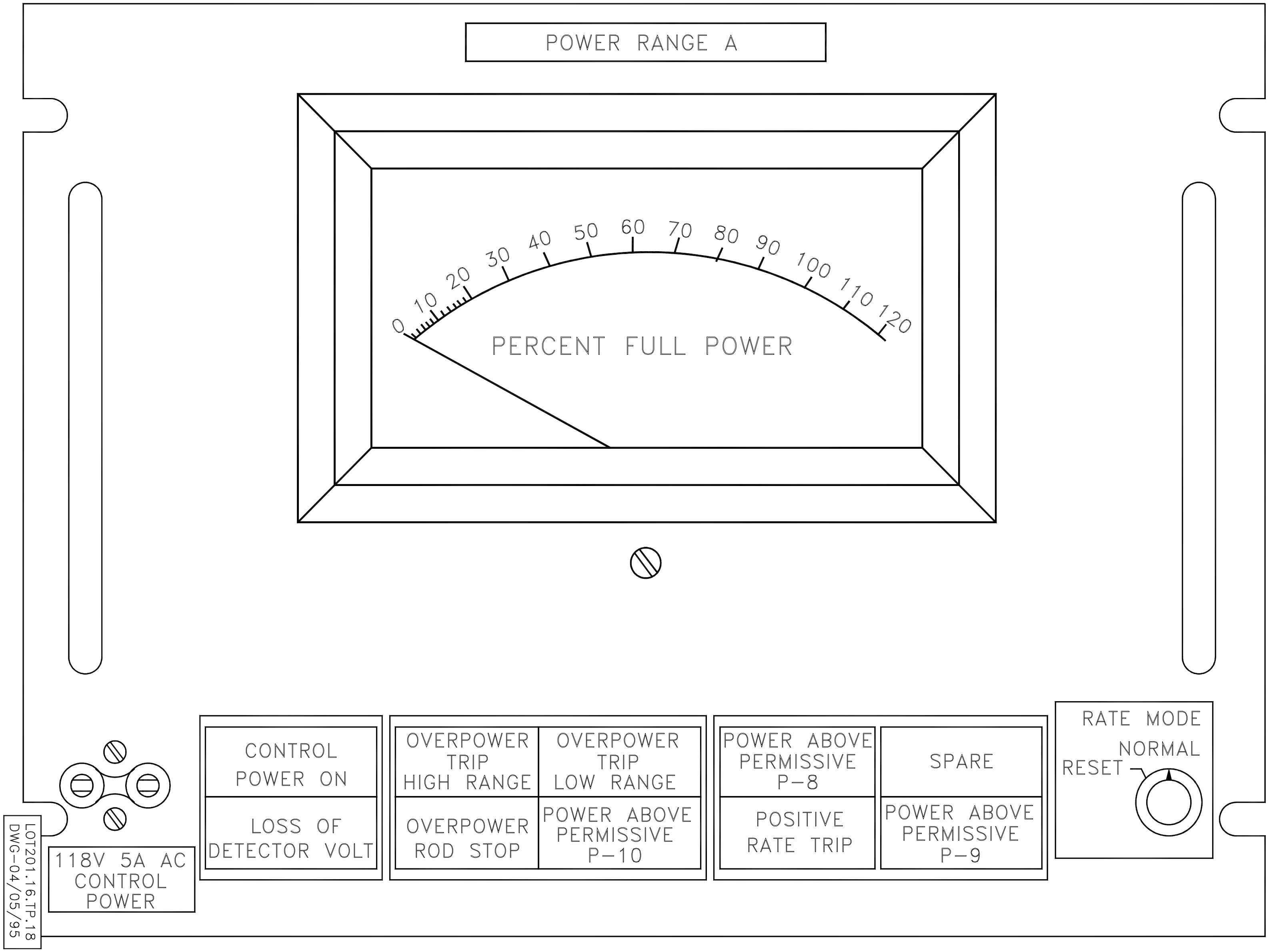


LOT201.16.TP.16
DWG - 11/28/02

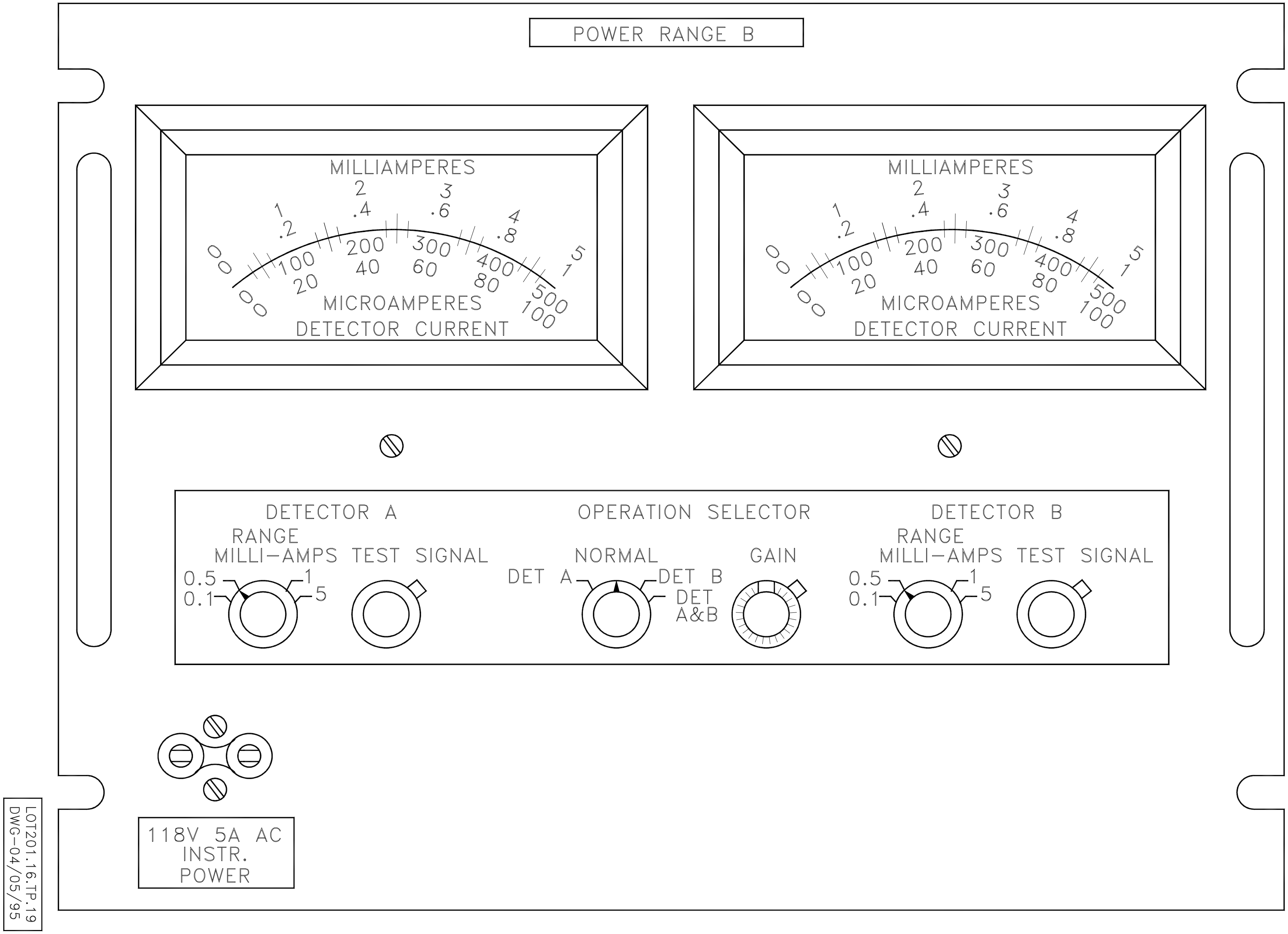
POWER RANGE INSTRUMENT

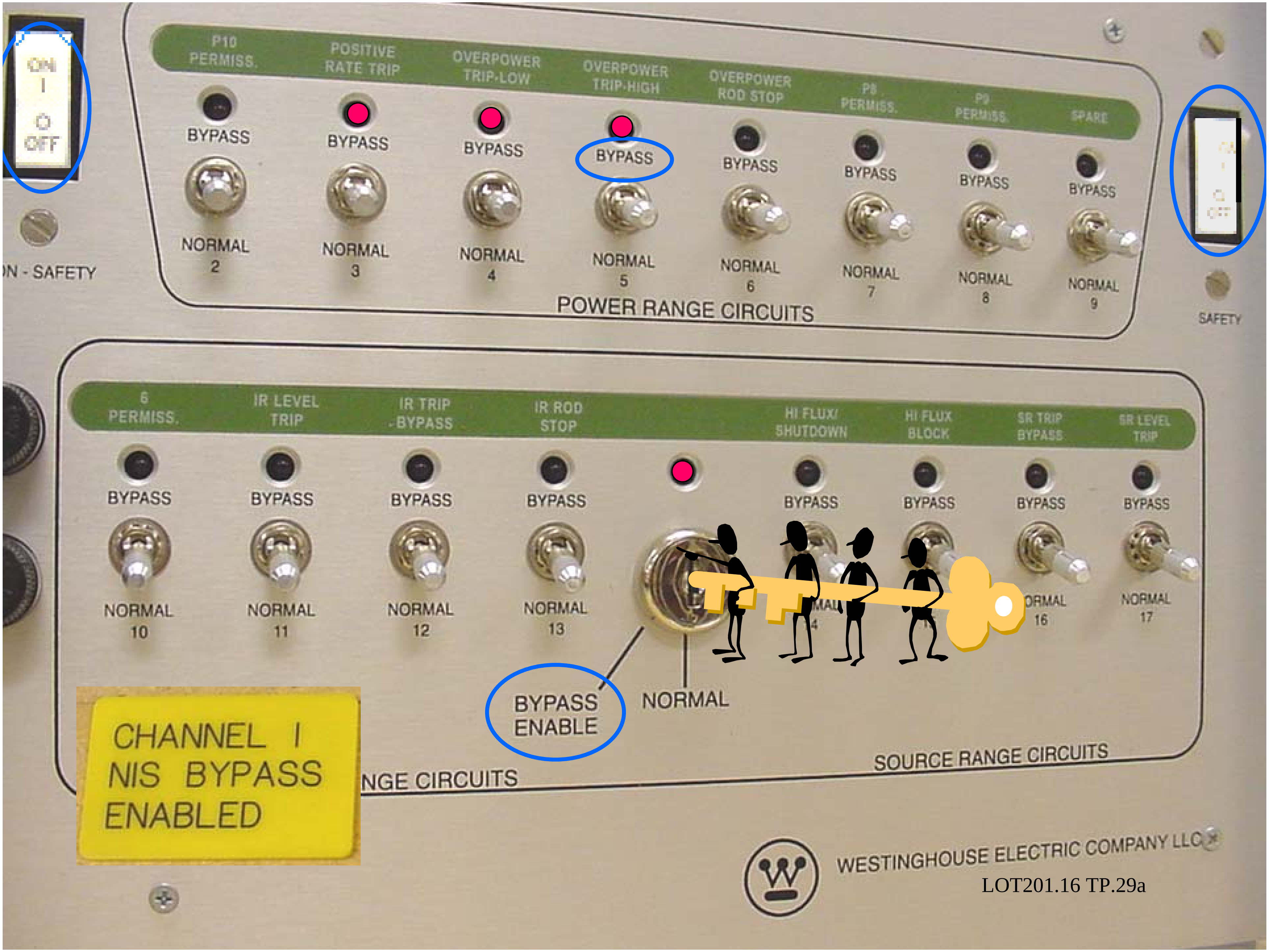


POWER RANGE DRAWER A



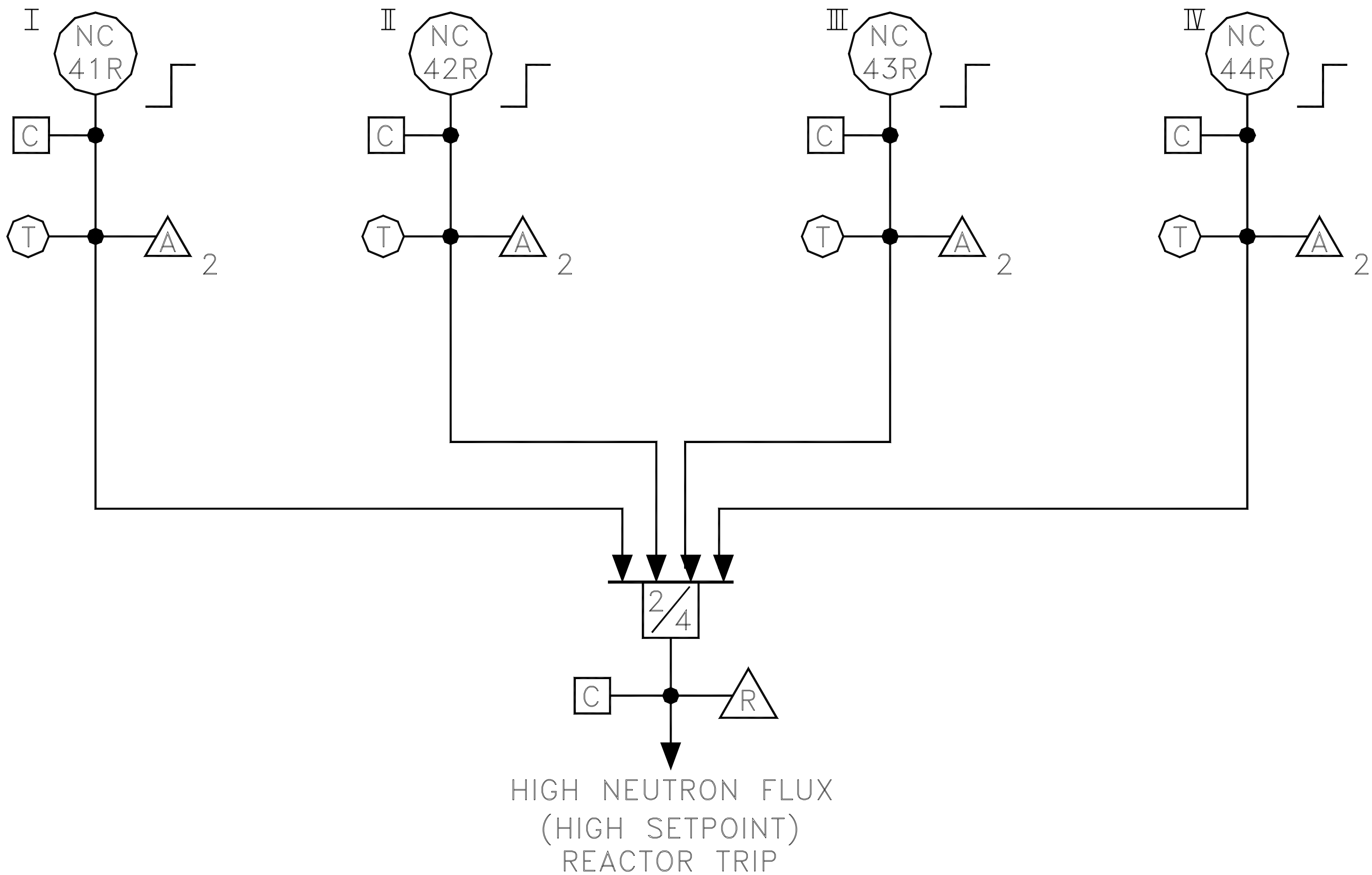
POWER RANGE DRAWER B





LOT201.16 TP.29a

POWER RANGE HI FLUX HI SETPOINT REACTOR TRIP



LOT201.16.TP.35
DWG - 04/07/95

Exam Bank No.: 2201 **Last used on an NRC exam:** Never

RO Sequence Number: 67

Given the following on Unit 1:

- A LOOP has occurred and all Emergency Diesel Generators (EDG) are running.
- A fire in the relay room has caused a Control Room evacuation and 0POP04-ZO-0001, Control Room Evacuation, is being performed.
- As secondary RO, you have performed the initial actions at ZLP-653 and ZLP-700 in Train A Switchgear Room per 0POP04-ZO-0001 to transfer equipment control to the local panels.

Which of the following describes actions associated with Essential Cooling Water (ECW) performed at ZLP-653 and 700 and the resultant affects on ESF Diesel #11 operation?

	Action(s)	Affects
A.	Transferred controls for ECW Pump A and discharge valve	ECW Pump A remains running and the discharge valve remains open with no affect on the operation of ESF DG #11
B.	Transferred control of ECW Pump A only	ECW Pump A remains running with no affect on the operation of ESF DG #11
C.	Transferred controls for ECW Pump A and discharge valve	ECW Pump A trips and the discharge valve remains open requiring the pump to be restarted to restore cooling to the DG.
D.	Transferred control of ECW Pump A only	ECW Pump A trips requiring the pump to be restarted to restore cooling to the DG.

Answer: B Transferred control of ECW Pump A only; ECW Pump A remains running with no affect on the operation of ESF DG #11

Exam Bank No.: 2201

K/A Catalog Number: 064 G2.4.34 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.2 **10CFR Reference:** 55.41(b)(10)

Emergency Diesel Generator: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

STP Lesson: LOT 201.13 **Objective Number:** 91193

LIST all automatic functions, switch locations, switch positions, annunciators (and where indicated), local/remote functions, interlocks and permissive for the following:
A. ECW Traveling Screens
B. ECW Screen Wash Booster Pumps
C. ECW Screen Wash Valves
D. ECW Strainers
E. ECW Pumps and Motors
F. ECW Discharge Valves
G. ECW Sump
H. ECW Blowdown Valve
I. ECW Sump Pump and Motor

Reference: LOT201.13 PowerPoint slides 40 & 44

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Action is credible because it would be reasonable to think the valve controls would be located with the pump controls (such as with the CCW pumps).

B: CORRECT: The secondary RO only transfers control of the ECW pump (valve control is transferred at the ECW structure). When control is transferred, equipment status is not affected.

C: INCORRECT: Action is credible because it would be reasonable to think the valve controls would be located with the pump controls (such as with the CCW pumps). Affect is credible because the transfer operation physically moves control of the equipment from one location to another which could result in the loss of the run signal and stop the pump.

D: INCORRECT: Affect is credible because the transfer operation physically moves control of the equipment from one location to another which could result in the loss of the run signal and stop the pump.

Question Level: F **Question Difficulty** 3

Justification:

Applicant requires a knowledge of local controls for ECW and the operational consequences of their use.

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)

DISCHARGE VALVE

- Transfer switch at the MCC
- "Local" control at the MCC prevents ECW Pump control from CR
- "Remote" control:

  Pump start: valve opens after 10 seconds
  Pump stop: valve closes after 2 minutes
- Status Monitoring alarm if: Valve fails to open

  Transfer switch in "LOCAL"
  Loss of control power
- Thermal overload computer alarm with indication on the MCC

Exam Bank No.: 2202 **Last used on an NRC exam:** Never

RO Sequence Number: 68

Which of the following describes the physical location of Core Exit Thermocouples and their relationship to the Reactor Coolant System?

A. Thermocouples are positioned just above the top of EACH Fuel Assembly and are used to determine Reactor Coolant System Subcooling.

B. Thermocouples are positioned just above the top of EACH Fuel Assembly and are used to determine Control Rod Insertion Limits.

C. Thermocouples are positioned just above the top of SELECTED Fuel Assemblies and are used to determine Control Rod Insertion Limits.

D. Thermocouples are positioned just above the top of SELECTED Fuel Assemblies and are used to determine Reactor Coolant System Subcooling.

Answer: D Thermocouples are positioned just above the top of SELECTED Fuel Assemblies and are used to determine Reactor Coolant System Subcooling.

Exam Bank No.: 2202

K/A Catalog Number: 017 K1.02 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(3)

Knowledge of the physical connections and/or cause-effect relationship between the ITM system and the following systems: RCS

STP Lesson: LOT 201.17 **Objective Number:** 91337

DESCRIBE the operation of the Incore Thermocouples.

Reference: LOT 201.17 lesson on Incore Nuclear Instrumentation.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because it would be reasonable to believe temperature is monitored at the exit of all assemblies.

B: INCORRECT: Number is credible because it would be reasonable to believe temperature is monitored at the exit of all assemblies. Useage is credible because the calculation for Control Rod Insertion Limits has an RCS temperature input, but it is core delta-T.

C: INCORRECT: Useage is credible because the calculation for Control Rod Insertion Limits has an RCS temperature input, but it is core delta-T.

D: CORRECT: There are 50 CETs placed just above selected Fuel Assemblies. They measure exit temperature of the core and feed into the calculation for RCS subcooling.

Question Level: F **Question Difficulty** 3

Justification:

The Reactor Operator needs to have fundamental knowledge of the operation of the core exit thermocouples.

LEAK DETECTION SYSTEM

The leak detection system consists of a drain header connecting the 10-path transfer devices, a pressure switch, drainage solenoid valve, an alarm light and reset pushbutton mounted on the distribution panel in the control room. Liquid collecting in a 10-path transfer due to a leak will cause the water level to rise in the drain header and thus actuate the pressure switch. The switch will then energize the audible alarm and the alarm light on the distribution panel while energizing the drainage solenoid to dump the water to the plant drain. The alarm is acknowledged by pressing the lighted reset pushbutton. This silences the audible alarm. When the water level in the drain header decreases below the pressure switch setting the alarm light goes out, the solenoid valve closes and the system returns to normal.

If a reactor coolant leak should develop in any of the incore thimbles, it may be detected by the leak detection system or by indicated abnormal radiation levels within the plant containment. Also difficulty of detector insertion may indicate a leaky thimble. Once such a leak has been detected, it should be possible to determine which thimble is faulty by visual observation at the path indicator switches (after removal of the covers). If it is a small leak, it may be possible to determine which thimble is affected by lightly touching the tubes between the seal plate and the movable frame to find the one having the highest temperature.

Proof of a small leak can best be done by inserting a dummy cable manually into the suspected thimble. If water is collected, it can be chemically analyzed to determine if it is reactor coolant. After identification of the leaking thimble, a small leak can be isolated during either a hot or cold shutdown by capping the appropriate thimble. A large leak would probably require cold shutdown for access to the thimble.

INCORE THERMOCOUPLE SYSTEM

Thermocouple

The Thermocouple System utilizes 50 thermocouples, positioned to measure fuel assembly coolant outlet temperature at preselected core locations. The thermocouples are the chromel-alumel type and have an accuracy of ±2°F. The thermocouple system is class IE, is divided into two redundant, independent and separate groups, Train A and Train C. (CET – 18 in Unit 1 is abandoned in place per DCP 06-677)

Thermocouple Routing and Seal Assemblies

Each thermocouple is 1/8-inch (nominal) diameter, stainless steel sheathed, aluminum oxide insulated, with the trailing end terminated in a male thermocouple connector. The thermoelectric characteristics conform to the K calibration curve within ±2° from zero to 530°F and within ±3/8 percent of point from 530°F to 700°F. Each thermocouple is supplied to the specific length required for its assigned location.

The sheaths, which are removable, are routed in guide tubes which position the thermocouple end at the selected core location. The guide tubes extend the entire distance from the core location to the seal assemblies.

Exam Bank No.: 2203 **Last used on an NRC exam:** Never

RO Sequence Number: 69

Unit 1 has just tripped from 100% Power.

The crew tripped the Reactor when it was determined that Control Rods were stepping out in an uncontrolled manner.

Given this condition, Deaerator (DA) level is expected to be…

A. trending up. Open DA high level dump bypass valves to prevent DA level from trending above 80%.

B. trending up. Close Condensate to DA inlet valves, CD-MOV-0574 & 0575 to prevent DA level from trending above 80%.

C. trending down. Start available Condensate Pumps to prevent DA level from trending below 30%.

D. trending down. Stop all Feedwater Booster Pumps to prevent DA level from trending below 30%.

Answer: A trending up. Open DA high level dump bypass valves to prevent DA level from trending above 80%.

Exam Bank No.: 2203

K/A Catalog Number: 045 A1.06 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with the MT/G system controls including: Expected response of secondary plant parameters following T/G trip.

STP Lesson: LOT 504.06 **Objective Number:** 81674

Given a step, note, or caution from 0POP05-EO-ES01, STATE/IDENTIFY the basis for the step, note or caution and the basis for the action to include the action itself, its purpose and result.

Reference: LOT 504.06 lesson on 0POP05-EO-ES01, Reactor Trip Response

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: DA level would trend up after a main turbine/generator trip with out complications on the secondary. Due to water entering the DA after the trip from sources other than the condensate inlet (e.g. condensate vent condenser) opening the DA high level dump bypass valves will be effective in controlling the high water level expected in the DA. Keeping DA level below 80% will prevent adverse effects on secondary piping.

B: INCORRECT: Credible because it is reasonable to think that closing the inlet MOVs would prevent water from entering the DA, however there are sources that bypass these valves.

C: INCORRECT: Trend is credible since several sources of water (MSDT discharge, FWH #11 drains) are no longer present. Action is credible because condensate pumps supply water to the DA and starting additional pumps would add additional water.

D: INCORRECT: Trend is credible since several sources of water (MSDT discharge, FWH #11 drains) are no longer present. Action is credible because the booster pumps remove water from the DA and the action would stop the removal.

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator needs to have knowledge of the response of the secondary plant after a main turbine/generator trip and be able to evaluate that the given condition would not complicate the secondary response to the main turbine/generator trip.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- o **This Step is performed to prevent adverse effects on secondary piping from high Deaerator level GREATER THAN 80% level.**
- o **Approximately three (3) turns open on each Deaerator High Level Dump Bypass Valve is a good starting point to maintain Deaerator level.**

___11 **MAINTAIN Deaerator Level:**

___a. VERIFY Deaerator Level Control Valve - MAINTAINING LEVEL

a. Manually CONTROL Deaerator Level Control Valve to maintain Deaerator Level.

___b. DISPATCH Operator To Throttle Open Deaerator High Level Dump Bypass Valves To Maintain Deaerator Level

(29 ft TGB S of CNDSR #13/23)

o "1(2)-FW-0486"
"DEAERATOR STORAGE TANK #2"
"HIGH LEVEL DUMP BYPASS VALVE"

o "1(2)-FW-0487"
"DEAERATOR STORAGE TANK #1"
"HIGH LEVEL DUMP BYPASS VALVE"

Exam Bank No.: 2204 **Last used on an NRC exam:** Never

RO Sequence Number: 70

Unit 1 is at 100% Power.

A Condensate Pump trips and the Standby Condensate Pump will not start leaving just one Condensate Pump running.

Which of the following components is affected to the extent that Operator Action is required to maintain proper operation?

Seal Water to the…

A. Low Pressure Heater Drip Pumps

B. Feedwater Booster Pumps

C. Condensate Pumps

D. Turbine Driven Steam Generator Feed Pumps

Answer: D Turbine Driven Steam Generator Feed Pumps

Exam Bank No.: 2204

K/A Catalog Number: 056 K1.03 **Tier:** 2 **Group/Category:** 2

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(7)

Knowledge of the physical connections and/or cause-effect relationship between the Condensate system and the following systems: Main Feedwater

STP Lesson: LOT 202.10 **Objective Number:** 40110

LIST all the systems that interface with the Condensate System and state the function of each interface.

Reference: LOT 202.10 lesson on the Condensate System and 0POP04-CD-0001, Loss of Condensate Flow

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because condensate supplies seal water to these pumps, but at a lower pressure.

B: INCORRECT: Credible because condensate supplies seal water to these pumps, but at a lower pressure.

C: INCORRECT: Credible because condensate supplies seal water to these pumps, but at a lower pressure.

D: CORRECT: Under the given condition, condensate pressure would not be high enough to supply proper seal water flow to the SGFPTs and operator action would be required to raise condensate pressure (manually reduce flow to the DA).

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator must be able to evaluate the given condition to determine the effect on other interfacing systems.

Step 3.0 continued from previous page

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
		e. CHECK for other valves in the Condensate to Deaerator flowpath that may be isolated. Refer to Addendum 4, Condensate Flowpath To the DA .
		f. IF flowpath can NOT be re-established, THEN COMMENCE a turbine load reduction per Addendum 3, Turbine Load Reduction.
_____ **4.0**	**CHECK Any Of The Following Annunciators Lit:** • "SGFPT 11(21) SEAL WTR DP LO" Lampbox 6M03, Window E-5 • "SGFPT 12(22) SEAL WTR DP LO" Lampbox 6M04, Window E-1 • "SGFPT 13(23) SEAL WTR DP LO" Lampbox 6M04, Window E-5	GO TO Step 6.0
_____ **5.0**	**TAKE Manual Control Of The DEAER Level Control Valves AND THROTTLE Closed Until Condensate Header Pressure Is High Enough To Clear All "SGFPT SEAL WTR DP LO" Alarms {CP009} (Approximately 400 To 450 PSIG)**	

Exam Bank No.: 2205 **Last used on an NRC exam:** Never

RO Sequence Number: 71

A fault in the steam pressure transmitter for the controlling Steam Flow channel on SG 1A causes the pressure indication to drop 50 psig.

Which one of the following describes the Steam Generator Water Level Control System response to this fault?

Assume program Δp for SGFPT speed control does NOT change.

SG 1A Main Feedwater Regulation Valve will initially …

A. open to match feedwater flow with indicated steam flow. When stabilized, SG 1A level will be controlling on program.

B. open to match feedwater flow with indicated steam flow. When stabilized, SG 1A level will be controlling slightly higher than program.

C. close to match feedwater flow with indicated steam flow. When stabilized, SG 1A level will be controlling on program.

D. close to match feedwater flow with indicated steam flow. When stabilized, SG 1A level will be controlling slightly lower than program.

Answer: C close to match feedwater flow with indicated steam flow. When stabilized, SG 1A level will be controlling on program.

Exam Bank No.: 2205

K/A Catalog Number: 035 A3.01 **Tier:** 2 **Group/Category:** 2

RO Importance: 4.0 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the S/G including: S/G water level control

STP Lesson: LOT 202.15 **Objective Number:** 21005

IDENTIFY the level controller, the manual/auto station, all input signals to and all output signals from the SGWLCS. STATE how a change in each input signal will affect the position of the Main Feed Regulating Valves.

Reference: LOT202.15 handout page 8 & 9

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because steam flow is pressure compensated and if the applicant incorrectly believes the failure will cause indicated steam flow to rise, then the valve response would be correct.

B: INCORRECT: Valve response is credible because steam flow is pressure compensated and if the applicant incorrectly believes the failure will cause indicated steam flow to rise, then the valve response would be correct. Long term response is credible because the indicated response would be correct if the controller did not have an integral function.

C: CORRECT: A drop in indicated steam pressure will cause a drop in indicated steam flow which will result in the MFRV closing to reduce feed flow to match indicated steam flow. Since the controller has an integral function, level will ultimately be restored to program.

D: INCORRECT: Long term response is credible because the indicated response would be correct if the controller did not have an integral function.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must first determine how the pressure change will affect steam flow, then apply the change in steam flow to the level control system to determine initial response. A knowledge of the type of controller (PI) is then needed to determine long term response of the affected SG level.

The large gain of 2.5 to 1 in the P+I level controller ensures that the control system is "level dominate" and will attempt to always return indicated level to setpoint. However, when a level instrument fails, the indicated level does not show actual level and the control system will respond to the failed instrument, driving actual level away from setpoint regardless of the magnitude of the instrument failure.

Anytime actual feed flow is changed, the feed-to-steam DP input to the SGFP Master Speed Controller changes. If a controlling level channel fails high or low, the actual feed flow and the feed/steam DP is decreased or increased accordingly. This will make the SGFPs speed up or slow down to recover the system DP to the programmed setpoint and may help in slowing the actual SG level change; however, the constant level error from the failed channel will continually attempt to close or open the MFRVs.

6.2 STEAM FLOW FAILURE

6.2.1 Steam Flow Channel Failing LOW

A controlling steam flow channel failing low from 100% power will initially result in a summed feed/steam "flow error" of 100%. This large "flow error" will rapidly decrease feedwater flow because the MFRV receives a signal to go closed to about 40% demand (a 0.6% proportional gain between "flow error" and "level error output" signal). The MFRV will slowly continue to close further as long as there is a difference between "flow error" and "level error".

As the MFRV closes, feedwater flow rapidly decreases. Actual SG level drops and now a level error is developed. Again, a proportional gain of 2.5 is applied on this level error to make this a level dominant system. When the "level error output" signal is large enough to overcome the "flow error", the MFRV will start to come open and increase feed flow in an attempt to restore actual SG level to program. However, with the Steam Flow channel still failed low, as soon as feed flow increases, the "flow error" gets larger and starts to counteract the "level error output" signal. From this point, it's a race to the SG LO LO level trip setpoint.

At the time of the steam flow channel failing low the total steam flow signal decreases resulting in a lower DP setpoint for the SGFP Master Controller. The SGFPs will eventually control to a lower DP setpoint. As the flow control valve closes, feedwater flow rapidly decreases, SG level drops and a level error is developed. A proportional gain of 2.5 is applied on this level error to make this a level dominate system.

An additional item to consider is the SGFPT speed control system. At the time of the steam flow channel failing low, the total steam flow signal decreases resulting in a lower DP setpoint for the SGFP Master Controller. The SGFPs will eventually control to a lower DP setpoint.

For example, with the plant at 100% power and a Steam Flow channel failed low, a 100% flow error is produced. As the MFRV closes, the actual feedwater flow decreases and the "flow error" rapidly decreases from 100%.

Once the "flow error" signal reaches <50%, for example, then it would take a level error signal of 20% to overcome the "flow error" (20% x 2.5 = 50% so, 50.7% actual SG level). However, once actual feed flow starts to increase, the "flow error" would get larger, and once again, close the MFRV. With the plant at 100% power, it would take a 40% level error signal to restore 100% feed flow (40% x 2.5 = 100% so, 30.7% actual SG level) With the SG LO LO level setpoint of 20 %, this means that SG level could theoretically be restored with no operator action.

This will happen on a controlling steam flow channel failing <u>low</u> when >50% power. On steam flow channel low failures when <50% power, the response is not as drastic and the resulting level error output" signal can overcome the "flow error" and will respond to return the SG level to setpoint. As long as the level deviates from setpoint, the P+I level controller will generate a "level error output" signal on an 1800 second reset time constant, allowing level to slowing return to setpoint with minimal overshoot.

6.2.2 Steam Flow Channel Failing HIGH

The feedwater and steam flow transmitters are calibrated for 0 to 118% flow. A controlling steam flow channel failing high from 100% results in an 18% "flow error" that will initially open the MFRV an additional 11% demand and then slowly continue to open the valve until the "level error output" signal becomes strong enough to start closing the valve. When the steam flow channel failed high the total steam flow signal increases and results in a higher DP setpoint for the SGFP Master Speed Controller. The SGFPs will eventually be controlled at a higher DP setpoint.

As actual feedwater flow increases, the SG level increases and a "level error output" signal is developed. When actual level increases > 7.2% (7.2% x 2.5 = 18), a "level error output" signal will be generated that is strong enough to overcome the 18% flow error. Additionally, with feedwater flow increasing, the "flow error" is lower and the level may not have to increase as far above setpoint before the generated "level error output" signal will start to close the MFRV. Because the "flow error" is < the level controller's "level error output" signal, it can compensate to bring feedwater flow back to 100% and slowly return level to setpoint.

If a controlling steam flow channel fails high at $\leq$ 68% power (a $\geq$ 50% "flow error"), the level controller's "level error output" signal

Exam Bank No.: 2209 **Last used on an NRC exam:** Never

RO Sequence Number: 72

Given the following:

- During core off load it is reported by the refueling crew that SFP and Rx cavity level is lowering.
- Plant personnel report water coming out of an open Steam Generator manway (nozzle dam failure)
- Control Room operators enter 0POP04-FC-0002, Refueling LOCA.
- A Plant Operator is dispatched to close the FUEL TRANSFER TUBE GATE VALVE per 0POP04-FC-0002.

Which of the following describes how the plant operator will close the valve and the resultant effect of the valve closure?

	Valve Closure	Effect
A.	Closed by manually turning local hand wheel	SFP level stabilized, ICSA and Rx Cavity continue to lower
B.	Closed by manually turning local hand wheel	SFP and ICSA level stabilized, Rx Cavity level continues to lower
C.	Closed by placing local Handswitch to CLOSE	SFP level stabilized, ICSA and Rx Cavity continue to lower
D.	Closed by placing local Handswitch to CLOSE	SFP and ICSA level stabilized, Rx Cavity level continues to lower

Answer: A closed by manually turning hand wheel; SFP level stabilized, ICSA and Rx Cavity continue to lower

Exam Bank No.: 2209

K/A Catalog Number: 033 G2.4.35 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.8 **10CFR Reference:** 55.41(b)(10)

Spent Fuel Pool Cooling: Knowledge of local auxilary operator task during an emergency and resultant operational effects.

STP Lesson: LOT 505.01 **Objective Number:** 38635

Given an abnormal operating event, PREDICT the symptoms expected to occur in accordance with the appropriate off normal operating procedure.

Reference: 0POP04-FC-0002 rev 14 and 0POP08-FH-0003 rev 32

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT - Fuel transfer canal gate valve is a manually operated valve which isolates the SFP from the ICSA and the Rx Cavity. With a leak inside containment, the ICSA and Rx cavity will continue to lower.

B: INCORRECT - Credible because there is also an ICSA gate, which if closed in this situation would produce the given indication.

C: INCORRECT - Credible because many large valves are motor operated (this one is not).

D: INCORRECT - Valve closure is credible because many large valves are motor operated (this one is not). Effect is credible because there is also an ICSA gate, which if closed in this situation would produce the given indication.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must have knowledge of and recall that the valve is manually operated and determine leak location based on the given information which will lead to a conclusion for overall plant status.

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____	**6.0**	**SUSPEND Movement Of Loads Over The SFP**	
_____	**7.0**	**CHECK Fuel Transfer Cart – AT SFP SIDE WITH UPENDER DOWN**	PLACE fuel transfer cart on the SFP side with the upender down.
_____	**8.0**	**CHECK 1(2)-FH-0001, "FUEL TRANSFER TUBE GATE VALVE" – CLOSED**	CLOSE 1(2)-FH-0001, "FUEL TRANSFER TUBE GATE VALVE"
_____	**9.0**	**CHECK Reactor Internals Movement – IN PROGRESS**	GO TO Step 11.0.
_____	**10.0**	**SECURE Reactor Internals In The Desired Storage Area**	
_____	**11.0**	**PERFORM The Following:** _____ **a. NOTIFY Health Physics of the current plant conditions** _____ **b. INSTRUCT Health Physics to perform a radiation survey of the SFP and RCB areas** _____ **c. ENSURE Appropriate actions are taken based on the results of the survey**	

8.4.10 OPEN the following power supply breakers for the RCB Control Console: [35 ft EAB Pen Space MCC 1K1(2K1)]

8.4.10.1 "1K1(2K1)/A3R RX SIDE FUEL HANDLING CONT PNL ZLP-111" _____

8.4.10.2 "1K1(2K1)/D4L B/U BKR-RX SIDE FUEL HANDLING CONT PNL ZLP-111" _____

CAUTION

- The Operator closing Fuel Transfer Tube Gate Valve SHALL be aware it takes approximately 109 turns to close gate valve from full open position.
- Closing “1(2)-FH-0001 FUEL TRANSFER SYSTEM TRANSFER TUBE ISOLATION VALVE” more than 109 turns may result in damage to the valve.
- IF “1(2)-FH-0001 FUEL TRANSFER SYSTEM TRANSFER TUBE ISOLATION VALVE” is turned approximately 109 turns in closed direction AND position indicator does NOT indicate closed, THEN valve SHALL **NOT** be closed any further until difference is resolved. (Reference 2.9)

8.4.11 CLOSE "1(2)-FH-0001 FUEL TRANSFER SYSTEM TRANSFER TUBE ISOLATION VALVE". (68 ft FHB N End of Transfer Canal) _____

8.4.12 NOTIFY Maintenance to close Fuel Transfer Quick Opening Hatch. _____

8.4.13 NOTIFY Health Physics that Fuel Transfer Operations through transfer canal have been secured. _____

Exam Bank No.: 2206 **Last used on an NRC exam:** Never

RO Sequence Number: 73

Unit 2 tripped from 100% Power due to a LOCKOUT on the Unit Aux Transformer.

Currently the Crew is performing 0POP05-EO-ES01, Reactor Trip Response, and is checking to see if Reactor Coolant Pumps can be started.

Main Steam has been isolated.

The following temperatures and pressures are reported by the Reactor Operator:

• PZR Pressure	1925 psig and lowering
• CET Temperature	605ºF and rising
• S/G Pressures	1200 psig and lowering
• RCS Hot Leg Temperatures	600 ºF and rising
• RCS Cold Leg Temperatures	595 ºF and rising
• RCS subcooling	27 ºF and lowering

Which of the following describes the action (and basis) the Crew should perform next?

A. Raise Steam Dumping Rate to aid in establishing natural circulation.

B. Raise Aux Feedwater Flow to aid in establishing natural circulation.

C. Initiate Safety Injection due to loss of subcooling.

D. Initiate Safety Injection to aid in identifying a faulted S/G.

Answer: C Initiate Safety Injection due to loss of subcooling.

Exam Bank No.: 2206

K/A Catalog Number: G2.1.7 **Tier:** 3 **Group/Category:** 1

RO Importance: 4.4 **10CFR Reference:** 55.41(b)(5)

Ability to evaluate plant performance and make operational judgements based on operating characteristics, reactor behavior, and instrument interpretation.

STP Lesson: LOT 504.06 **Objective Number:** 81674

Given a step, note, or caution from 0POP05-EO-ES01, STATE/IDENTIFY the basis for the step, note or caution and the basis for the action to include the action itself, its purpose and result.

Reference: LOT 504.06 lesson on 0POP05-EO-ES01, Reactor Trip Response (CIP) and Steam Tables

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because the given conditions indicate natural circulation is not occurring and raising steaming rate would be appropriate to help establish natural circulation.

B: INCORRECT: Credible because the given conditions indicate natural circulation is not occurring and raising Aux Feedwater flow could help establish natural circulation.

C: CORRECT: With RCS Subcooling at 27 degrees F, initiation of SI is required because it is below the required limit of 35 degrees F. (See procedure CIP)

D: INCORRECT: Credible because an SI is required, however it would not assist in identifying a faulted generator (a main steam isolation would) and there are no given indications of a faulted generator (pressures are lowering but it is due to loss of natural circulation).

Question Level: H **Question Difficulty** 3

Justification:

The Reactor Operator needs to be able to evaluate the given condition and determine the appropriate action.

CONDITIONAL INFORMATION PAGE

SI ACTUATION CRITERIA

IF EITHER condition listed below occurs, THEN ACTUATE SI and GO TO 0POP05-EO-EO00, REACTOR TRIP OR SAFETY INJECTION, Step 1.

- o RCS subcooling based on core exit T/Cs - LESS THAN 35°F.
- o Pressurizer level - LESS THAN 8%.

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.

EO00 TRANSITION CRITERIA

IF an SI actuation occurs during this procedure, THEN GO TO 0POP05-EO-EO00, REACTOR TRIP OR SAFETY INJECTION, Step 1.

RCFC COOLING CRITERIA

RESTORE CCW to RCFCs within 30 minutes after a LOOP signal.

SEQUENCER LOADING VERIFICATION

IF a LOOP has occurred, THEN PERFORM Addendum 5, Sequencer Loading Verification - Mode II.

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.)

Exam Bank No.: 2071 **Last used on an NRC exam:** Never

RO Sequence Number: 74

Given the following:

- New Fuel receipt is in progress in the Fuel Handling Building.
- A New Fuel Assembly is dropped resulting in a breach of the Fuel Cladding.

The resulting radiation hazard is primarily…

A. EXTERNAL exposure due to the presence of neutron radiation.

B. EXTERNAL exposure due to the presence of alpha radiation.

C. INTERNAL exposure due to the presence of neutron radiation.

D. INTERNAL exposure due to the presence of alpha radiation.

Answer: D INTERNAL exposure due to the presence of alpha radiation.

Exam Bank No.: 2071

K/A Catalog Number: G2.3.14 **Tier:** 3 **Group/Category:** 3

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(12)

Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

STP Lesson: LOT 103.02 **Objective Number:** N91217

Contrast the effects of external and internal exposure.

Reference: LOT 103.04 student handout page 23

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Credible because neutron radiation is generally associated with nuclear fuel, more specifically it should be spent nuclear fuel. The hazard is credible because if there were neutrons, the hazard would be primarily external as stated.

B: INCORRECT: Credible because all other forms of radiation have sufficient penetrating power to be an external hazard, however Alpha particles travel only a short distance due to their ionizing potential and can easily be shielded by a person's skin.

C: INCORRECT: Credible because neutron radiation is generally associated with nuclear fuel, more specifically it should be spent nuclear fuel.

D: CORRECT: Alpha radiation is the primary hazard for new fuel due to the decay of the uranimum. Alpha particles travel only a short distance due to their ionizing potential and can easily be shielded by a person's skin.

Question Level: F **Question Difficulty** 3

Justification:

Applicant must know the radiation/contamination hazard for new fuel and that alpha particles are primarily an internal radiation hazard.

Concrete 0.089 cm^{-1}

Rule of Thumb: Neutron dose rate in water.

The tenth-thickness concept can also be applied to neutron dose rate attenuation. The rule of thumb is: 10 inches of water or polyethylene is necessary to reduce the neutron dose rate by a factor of 10.

Example:

Calculate the tenth value layer of water for fast neutrons given $\Sigma_r(H_20) = 0.103 \ cm^{-1}$.

Solution:

$$I = I_o e^{-\Sigma_r x}$$
$$I = \frac{1}{10} I_o$$

$$\frac{1}{10} I_o = I_o e^{-\Sigma_r x_{1/10}}$$
$$\frac{1}{10} = e^{-(0.103cm^{-1})x_{1/10}}$$
$$\ln\frac{1}{10} = (0.103cm^{-1})x_{1/10}$$
$$x_{1/10} = \frac{\ln\frac{1}{10}}{-0.103cm^{-1}}$$
$$x_{1/10} = 22.4cm = 8.8inches$$

Alpha and Beta Attenuation:

Since alpha and beta radiation have relatively low penetrating power (because of their high ionizing potential) they are primarily an internal radiation hazard.

Alpha particles lose energy rapidly in any medium because of high specific ionization (large size and charge). Alpha particles normally produced by fission are contained within the fuel elements and furthermore can be stopped by a sheet of paper. The outer layer of skin will absorb alpha particles up to 7.5 MeV.

Betas are usually absorbed by material containing the radioactive source or by any shielding employed to reduce gamma levels. Beta particles, have a lower specific ionization, therefore their penetration into any absorber will be much greater that of an alpha particle. Beta is considered a slight external hazard, since a 70 KeV beta will penetrate the skin. It is primarily a hazard to the lens of the eyes.

Exam Bank No.: 1830 **Last used on an NRC exam:** 2009

RO Sequence Number: 75

To satisfy the Reactor Coolant System Pressure SAFETY LIMIT, Reactor Coolant System pressure cannot exceed _______ psig.

A. 2380

B. 2485

C. 2735

D. 3110

Answer: C 2735

Exam Bank No.: 1830

K/A Catalog Number: G2.2.22 **Tier:** 3 **Group/Category:** 2

RO Importance: 4.0 **10CFR Reference:** 55.41(b)(5)

Knowledge of limiting conditions for operations and safety limits.

STP Lesson: LOT 201.02 **Objective Number:** 92102

(RCS) 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 adminstrative requirements affected, limitations, major time frames involved, major surveillance in order to comply, and the bases for the specification.

Reference: Safety Limit 2.1.2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Credible because 2380 is an RCS setpoint (high pressure trip setpoint).

B: INCORRECT: Credible because 2485 is an RCS setpoint (Pzr safety lift setpoint).

C: CORRECT: Per Safety Limit 2.1.2, RCS pressure must not exceed 2735 psig.

D: INCORRECT: Credible because 3110 is an RCS setpoint (hydro criteria for the RCS).

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have a knowledge of the Safety Limits.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of THERMAL POWER, pressurizer pressure, and the highest operating loop coolant temperature (T_{avg}) shall not exceed the limits shown in the Core Operating Limits Report.

2.1.1.1 In MODES 1 and 2, the departure from nucleate boiling ratio (DNBR) shall be maintained $\geq$ 1.17 for the WRB-1 DNB correlation and $\geq$ 1.14 for the WRB-2M DNB correlation.

2.1.1.2 In MODES 1 and 2, the peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58 °F per 10,000 MWD/MTU of burnup.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of the highest operating loop average temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour, and comply with the requirements of 10 CFR 50.36(c)(1).

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2735 psig.

APPLICABILITY: MODES 1, 2, 3, 4, AND 5.

ACTION:

MODES 1 and 2:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour, and comply with the requirements of 10 CFR 50.36(c)(1).

MODES 3, 4 and 5:

Whenever the Reactor Coolant System pressure has exceeded 2735 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes, and comply with the requirements of 10 CFR 50.36(c)(1).

Exam Bank No.: 2181 **Last used on an NRC exam:** Never

SRO Sequence Number: 76

The following events occur simultaneously:

- A Fire in the Relay Room Requiring Fire Brigade Response
- A Large Main Steam Line Break in Containment

Which of the following procedures takes precedence?

A. 0POP05-EO-EO00, Reactor Trip or Safety Injection

B. 0POP05-EO-FRZ1, Response to High Containment Pressure

C. 0POP04-ZO-0008, Fire/Explosion

D. 0POP04-ZO-0001, Control Room Evacuation

Answer: D 0POP04-ZO-0001, Control Room Evacuation

Exam Bank No.: 2181

K/A Catalog Number: G2.4.16 **Tier:** 3 **Group/Category:** 4

SRO Importance: 4.4 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.

STP Lesson: LOT 504.04 **Objective Number:** 92283

Given a set of conditions and the occurrence of a Red, Orange, or Yellow path CSF, STATE the action required per 0POP01-ZA-0018, EOP Users Guide.

Reference: LOT 504.04 - 0POP01-ZA-0018, Emergency Operating Procedure User's Guide

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because this is the procedure normally entered first in the event of a steam break.

B: INCORRECT: Credible because a large steam break in containment will require performing this procedure.

C: INCORRECT: Credible because this procedure will be entered during a fire, but does not take precedence.

D: CORRECT: 0POP04-ZO-0001, Control Room Evacuation, takes precedence over all other EOPs and 0POP04-ZO-0008, Fire/Explosion or 0POP04-ZO-0009, Safe Shutdown Fire Response.

Question Level: F **Question Difficulty** 3

Justification:

The Unit Supervisor needs to have knowledge of the priority of procedure usage for a given condition.

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.)

Exam Bank No.: 2178 **Last used on an NRC exam:** Never

SRO Sequence Number: 77

Authorization has been given to allow work on the packing of a motor operated valve (MOV) using the valve backseat as the boundary.

In accordance with 0PGP03-ZO-ECO1A, Equipment Clearance Order Instructions, who was responsible for giving authorization to perform the work and how shall the authorization be documented?

	AUTHORIZATION	DOCUMENTATION
A	Plant Manager and Engineering Division Manager	In the ECO Notes section of the ECO Form
B	Plant Manager and Engineering Division Manager	In the General Information of the Shift Manager Shift Turnover Checklist
C	Unit Operations Manager and Maintenance Division Manager	In the General Information of the Shift Manager Shift Turnover Checklist
D	Unit Operations Manager and Maintenance Division Manager	In the ECO Notes section of the ECO Form

Answer: D Unit Operations Manager and Maintenance Division Manager. In the ECO Notes section of the ECO Form.

Exam Bank No.: 2178

K/A Catalog Number: G2.2.13 **Tier:** 3 **Group/Category:** 2

SRO Importance: 4.3 **10CFR Reference or SRO Objective:** 55.43(b)(3)

Knowledge of tagging and clearance procedures.

STP Lesson: LOT 802.31 **Objective Number:** SRO-1172

STATE the MOV Manual Seating Guidelines

Reference: LOT 507.01 - 0PGP03-ZO-ECO1A, Equipment Clearance Order Instructions, addendum 3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because at one time the Plant Manager was the approval authority for this and engineering is considered the equipment experts.

B: INCORRECT: Credible because at one time the Plant Manager was the approval authority for this and engineering is considered the equipment experts. Documentation is credible because the authorization could be listed on the Shift Manager Turnover Checklist, but there is no retention requirements for use of the checklist whereas there is retention requirements for an ECO.

C: INCORRECT: Documentation is credible because the authorization could be listed on the Shift Manager Turnover Checklist, but there is no retention requirements for use of the checklist whereas there is retention requirements for an ECO.

D: CORRECT: Lists the correct authorization and documentation requirements.

Question Level: F **Question Difficulty** 3

Justification:

SRO must have knowledge of authorization and administrative requirements of the ECO program.

CAUTION

- MOV’s should be manually seated (handwheel) with minimum applied force to prevent exceeding the compensating spring pack deflection.
- Valve wrenches SHALL NOT be used on MOV’s for manual seating. (Reference 2.2.24)
- Manually seated valves should be returned to normal position prior to a thermal cycle on the valve.
- For Technical Specification required cooldown, manually seated MOV’s SHALL be returned to normal position as soon as possible.
- Any safety-related MOV that is manually seated, SHALL be evaluated for operability by the Shift Manager/Unit Supervisor for entry into OAS.
- Declutching and manually closing an MOV is NOT the normal method of isolating equipment AND requires notification of the Field OR Unit Supervisor. This action will allow some MOV’s to OPEN due to system differential pressure AND requires the valve to be manually unseated prior to energizing the MOV motor upon restoration.
- Whenever possible, the MOV should be de-energized prior to manual operation.
- Concurrent manual and remote operation of an MOV SHALL NEVER be attempted.

1. Motor Operated Valves (MOV) Manipulation:

   1.1 MOVs should normally be opened or closed using the motor handswitch. <u>IF</u> the MOV has a handswitch in the Control Room, <u>THEN</u> the Control Room handswitch should be used for valve positioning.

   1.2 Any time an MOV must be declutched and manually closed for ECOs (to stop system leakage), the Field or Unit Supervisor will be informed and the applicable ECO noted. This is to ensure the MOV is manually unseated on ECO restoration <u>prior to</u> energizing the motor. Also see Chapter 9 of Conduct of Operations Manual for MOV manipulation.

2. The backseat of a valve may be used as a BOUNDARY for maintenance provided that:

   2.1 The applicable WORK DOCUMENT allows for maintenance on the valve on its backseat.

   2.2 The applicable Unit Operations Manager and the applicable Maintenance Division Manager or General Maintenance Supervisor have authorized performance of maintenance on the valve on the backseat. This authorization MAY be delivered verbally to the Shift Manager.

   2.3 This authorization, including the date, time and method (phone, email, etc.), SHALL be documented in the ECO Notes.

Exam Bank No.: 1627 **Last used on an NRC exam:** 2007

SRO Sequence Number: 78

Unit 2 is in Mode 6 when the following occurs:

- CNTMT NORMAL SUMP LVL HI-HI alarm actuates
- CNTMT SEC NORM SUMP LVL HI-HI alarm actuates
- Personnel in the Fuel Handling Building report lowering level in the Spent Fuel Pool

Based on this information, which of the following correctly identifies the location of the leak AND the procedure to be entered by the Unit Supervisor?

A. Spent Fuel Pool Cooling leak in the Fuel Handling Building; 0POP04-RC-0007, Mode 5 Or Mode 6 LOCA With The Reactor Vessel Head On

B. Spent Fuel Pool Cooling leak in the Fuel Handling Building; 0POP04-FC-0002, Refueling LOCA

C. Residual Heat Removal System leak in containment; 0POP04-RC-0007, Mode 5 Or Mode 6 LOCA With The Reactor Vessel Head On

D. Residual Heat Removal System leak in containment; 0POP04-FC-0002, Refueling LOCA

Answer: D Residual Heat Removal System leak in containment; POP04-FC-0002, Refueling LOCA

Exam Bank No.: 1627

K/A Catalog Number: APE 025 AA2.03 **Tier:** 1 **Group/Category:** 1

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of RHR System: Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Increasing reactor building sump level

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

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

Reference: LOT 505.01 - Instruction on 0POP04-FC-0002, Refueling LOCA

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified From**

Distractor Justification

A: INCORRECT - Credible leak location because SFP level is lowering. Credible procedure because 0POP04-RC-0007 is used for a LOCA in Mode 6 (but only with the RX Head on). With level lowering in the SFP and levels rising in the RCB Sumps this would indicate that the RX Head is off and the SFP is tied to the RX Cavity through the Fuel Transfer Canal making 0POP04-FC-0002 the correct procedure.

B: INCORRECT - Credible leak location because SFP level is lowering.

C: INCORRECT - Credible because 0POP04-RC-0007 is used for a LOCA in Mode 6 but only with the RX Head on. With level lowering in the SFP and levels rising in the RCB Sumps this would indicate that the RX Head is off.

D: CORRECT - Sump alarms given are in the containment, SFP level is lowering because the transfer tube is open and the cavity flooded with the reactor head off. The correct procedure to use during a LOCA with the head off is POP04-FC-0002

Question Level: H **Question Difficulty** 3

Justification:

The applicant must determine that the leak is from RHR in containment (sump alarms) even though SFP level is lowering (which it will with an RHR leak while the transfer tube is open). The applicant must also have a knowledge of the entry conditions (Refueling LOCA vs Mode 5 or 6 LOCA) of the referenced procedures.

PURPOSE

This procedure provides guidelines for protection of the Reactor Core, Fuel Assemblies, and Personnel in the event of a LOCA with the Reactor Vessel Head off of the vessel.

SYMPTOMS OR ENTRY CONDITIONS

1. Any of the following annunciator alarms are possible symptoms:

• "CNTMT NORM SUMP LVL HI-HI"	Lampbox 5M03, Window D-8
• "CNTMT SEC NORM SUMP LVL HI-HI"	Lampbox 5M03, Window E-8
• "SFP WATER LVL HI/LO"	Lampbox 22M02, Window F-5
• "SFP TROUBLE"	Lampbox 22M02, Window F-6

2. Any of the following local indications are entry conditions:
   - SG Nozzle Dam failure.
   - Rising Reactor Containment Building (RCB) radiation levels (RM-11, RM-23).
   - Containment Ventilation Isolation (CVI) due to high radiation levels.
   - Rising Spent Fuel Pool (SFP) area radiation monitor levels (RT-8090 or RT-8091).
   - Fuel Handling Building Ventilation system shifted to Emergency Mode of operation due to high radiation levels.
   - Reactor Cavity or SFP water level lowering in an uncontrolled manner.

Exam Bank No.: 2154 **Last used on an NRC exam:** Never

SRO Sequence Number: 79

Unit 1 was operating at 100% power when an event occurred that tripped the reactor and initiated a Safety Injection.

The crew is performing 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Based on the following conditions of the Steam Generators and Containment;

Steam Generators	A	B	C	D
Pressure	1095 psig Slowly Lowering	1085 psig Slowly Lowering	1090 psig Slowly Lowering	1010 psig Slowly Lowering
Level	20% NR Slowly Rising	19% NR Slowly Rising	29% NR Stable	31% NR Slowly Lowering
AFW Flow	150 gpm	150 gpm	50 gpm	50 gpm

Containment	
Pressure	3.2 psig - Rising
Temperature	130ºF - Rising
Humidity	110ºF-dew point – Rising

Which of the following procedures should the Unit Supervisor perform next?

A. 0POP05-EO-EO20, Faulted Steam Generator Isolation

B. 0POP05-EO-EO30, Steam Generator Tube Rupture

C. 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant

D. 0POP05-EO-FRZ1, Response to High Containment Pressure

Answer: C 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant

Exam Bank No.: 2154

K/A Catalog Number: EPE 009 EA2.11 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.1 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to determine or interpret the following as they apply to a Small Break LOCA: Containment temperature, pressure and humidity

STP Lesson: LOT 504.05 **Objective Number:** 80474

From memory STATE/IDENTIFY how the RCS is checked to be intact per POP05-EO-EO00, Reactor Trip or Safety Injection.

Reference: LOT 504.05 - Procedure training on 0POP05-EO-EO00, Reactor Trip or Safety Injection

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because SG D pressure is lower than the other three with its level is also lowering and the given containment conditions could be caused by a steam leak inside the RCB. However this is normal for the given conditions due to the turbine driven aux feedwater pump that is connected to this steam generator.

B: INCORRECT: Credible because there are SG levels that are rising. However closer analysis of the given conditions reveals that the SGs with rising level are also being supplied with more AFW flow, which with no other indications should rule out tube leakage.

C: CORRECT: Containment conditions given are indicative of a SBLOCA or small steam break, however secondary indications do not support the steam break.

D: INCORRECT: Credible because containment pressure is elevated. However, it has not yet reached the level required for 0POP05-EO-FRZ1 entry.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate the given conditions to determine the event that has occurred and then determine which procedure is appropriate to transition to based on the event in progress.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___12	**CHECK If SG Secondary Pressure Boundary Intact:**	
	___a. CHECK pressures in all SGs - o CONTROLLED <u>OR</u> RISING o GREATER THAN CONTAINMENT PRESSURE	a. <u>IF</u> any faulted SG is <u>NOT</u> isolated, <u>AND</u> is <u>NOT</u> needed for RCS cooldown, <u>THEN</u> PERFORM the following: o GO TO 0POP05-EO-EO20, FAULTED STEAM GENERATOR ISOLATION, Step 1. o MONITOR Critical Safety Functions. o <u>WHEN</u> Addendum 5 of this procedure is complete, <u>THEN</u> Functional Restoration Procedures may be IMPLEMENTED. -------------------------------------
___13	**CHECK If SG Tubes Are Intact:** o Main steamline radiation - NORMAL o <u>IF</u> SG blowdown in service, <u>THEN</u> SG blowdown radiation - NORMAL o CARS pump radiation - NORMAL o NO SG level rising in an uncontrolled manner	GO TO 0POP05-EO-EO30, STEAM GENERATOR TUBE RUPTURE, Step 1. o MONITOR Critical Safety Functions. o <u>WHEN</u> Addendum 5 of this procedure is complete, <u>THEN</u> Functional Restoration Procedures may be IMPLEMENTED. -------------------------------------
___14	**CHECK If RCS Is Intact:** o Containment radiation - NORMAL o Containment pressure - NORMAL o Containment wide range water level - NORMAL	GO TO 0POP05-EO-EO10, LOSS OF REACTOR OR SECONDARY COOLANT, Step 1. o MONITOR Critical Safety Functions. o <u>WHEN</u> Addendum 5 of this procedure is complete, <u>THEN</u> Functional Restoration Procedures may be IMPLEMENTED. -------------------------------------

Exam Bank No.: 2155 **Last used on an NRC exam:** Never

SRO Sequence Number: 80

A reactor trip has occurred from 2% reactor power.

- The crew is performing Step 1 of 0POP05-EO-ES01, Reactor Trip Response
- Source Range Detectors have not energized yet.

Subsequently a Reactor Operator reports the following;

- Extended Range Startup Rate is 0.1 DPM.
- DRPI shows three control rods at 24 steps.

Which indication should the Unit Supervisor interpret as a priority condition and which procedure should be entered?

A. Extended Range Startup Rate is 0.1 DPM – Enter 0POP04-CV-0003, Emergency Boration

B. Extended Range Startup Rate is 0.1 DPM – Enter 0POP05-FO-FRS1, Response to Nuclear Power Generation – ATWS

C. DRPI indicates three control rods at 24 steps – Enter 0POP04-CV-0003, Emergency Boration

D. DRPI indicates three control rods at 24 steps – Enter 0POP05-FO-FRS1, Response to Nuclear Power Generation – ATWS

Answer: B Extended Range Startup Rate is 0.1 DPM - Enter 0POP05-FO-FRS1, Response to Nuclear Power Generation - ATWS

Exam Bank No.: 2155

K/A Catalog Number: EPE 029 G2.1.7 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.7 **10CFR Reference or SRO Objective:** 55.43(b)(5)

ATWS: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavor and instrument interpretation.

STP Lesson: LOT 504.28 **Objective Number:** 84506

STATE the basis for monitoring the conditions listed in the subcriticality safety function status tree associated with 0POP05-EO-FRS1.

Reference: LOT 504.28 Instruction on 0POP05-FO-FRS1, Response to Nuclear Power Generation - ATWS

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because the POP04 procedure will add boron to the RCS, however within the EOP network, using the FRP is the correct response.

B: CORRECT: A positive start up rate would be a priority condition because it is a true indication of core reactivity and entering the critical safety function procedure would be the correct path to take. 0POP05-EO-FRS1 would be entered with RX power greater than 5E-6% (Source Range NIs not energized yet = RX power of at least 1E-5% or greater) and start up rate greater than 0.

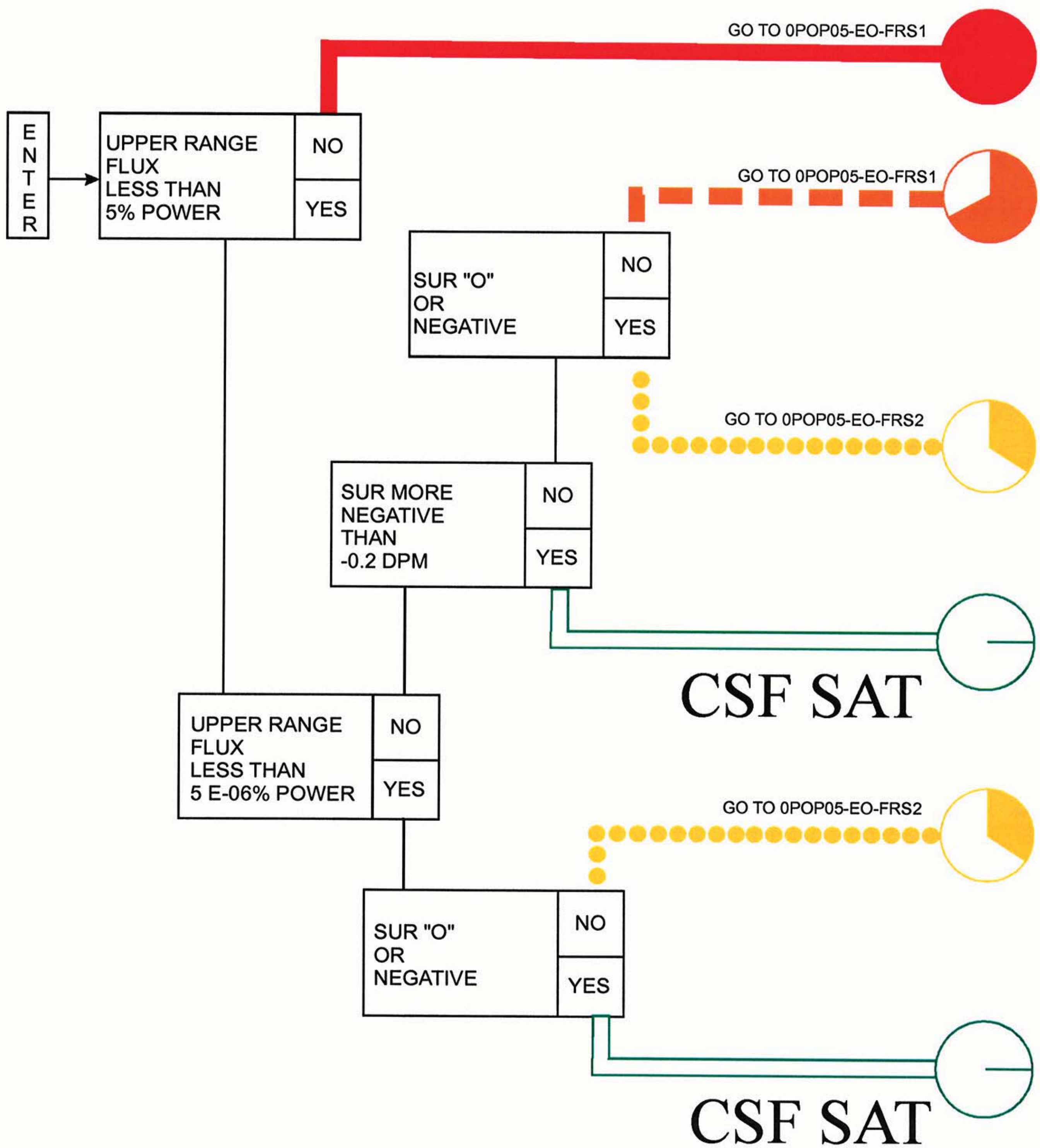
C: INCORRECT: Credible because DRPI is indication of rod position which directly affects core reactivity. The indicated procedure is credible because the POP04 procedure will add boron to the RCS, however within the EOP network, using the FRP is the correct response.

D: INCORRECT: Credible because DRPI is indication of rod position which directly affects core reactivity.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate the given instrument response and enter the appropriate procedure for the condition.


ENTER
UPPER RANGE FLUX LESS THAN 5% POWER
NO
YES
GO TO 0POP05-EO-FRS1
GO TO 0POP05-EO-FRS1
SUR "O" OR NEGATIVE
NO
YES
GO TO 0POP05-EO-FRS2
SUR MORE NEGATIVE THAN -0.2 DPM
NO
YES
CSF SAT
UPPER RANGE FLUX LESS THAN 5 E-06% POWER
NO
YES
GO TO 0POP05-EO-FRS2
SUR "O" OR NEGATIVE
NO
YES
CSF SAT

Exam Bank No.: 2158 **Last used on an NRC exam:** Never

SRO Sequence Number: 81

Unit 1 is at 100% Power and stable with all systems in a normal lineup.

An event occurs and an RO reports the following for Steam Generator levels:

- Steam Generator 'A' – 73% and rising.
- Steam Generator 'B' – 73% and rising.
- Steam Generator 'C' – 70% and stable.
- Steam Generator 'D' – 70% and stable.

Which of the following (1) describes a failure that could cause this event and (2) the procedure that has the appropriate actions to control Steam Generator levels?

A. (1) Loss of power to Feedwater Pump Discharge Pressure Transmitter PT-0558
(2) Enter 0POP04-VA-0001, Loss of 120 VAC Class Vital Distribution, to manually control SGFP Master Speed Controller.

B. (1) Loss of power to Feedwater Pump Discharge Pressure Transmitter PT-0558
(2) Enter 0POP04-FW-0002, Steam Generator Feed Pump Trip, to manually control SGFP Master Speed Controller.

C. (1) Loss of power to Class 1E 120 VAC DP-1201
(2) Enter 0POP04-VA-0001, Loss of 120 VAC Class Vital Distribution, to deselect affected channels controlling Steam Generator water level.

D. (1) Loss of power to Class 1E 120 VAC DP-1201
(2) Enter 0POP04-FW-0002, Steam Generator Feed Pump Trip, to deselect affected channels controlling Steam Generator water level.

Answer: C (1) Loss of power to Class 1E 120 VAC DP-1201
(2) Enter 0POP04-VA-0001, Loss of 120 VAC Class Vital Distribution, to deselect affected channels controlling Steam Generator water level.

Exam Bank No.: 2158

K/A Catalog Number: APE 057 G2.4.49 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.4 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Vital AC Instrument Bus: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

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

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

Reference: LOT 505.01 lesson on off normal procedure 0POP04-VA-0001, Loss of 120 VAC Class Vital Distribution.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible beacause a loss of power to PT-0558 would cause SG levels to rise, however all would rise. The procedure is credible because it contains steps for controlling SG levels, but would not be entered on a loss of power to PT-0558 (a non-class instrument).

B: INCORRECT: Credible beacause a loss of power to PT-0558 would cause SG levels to rise, however all would rise. The procedure is credible because it contains steps for responding to a malfunction of PT-0558.

C: CORRECT: With a normal lineup of SG level controls, a loss of DP-1201 would cause level to rise on SG A and B. 0POP04-VA-0001 has the required actions to conrol SG level in more than one SG and specifically has the extra action to deselect the affected channels.

D: INCORRECT: The procedure is credible because it contains steps for controlling SG levels under abnormal conditions (i.e. loss of PT-0557 or 0558, failure of master speed controller), but not this condition.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Suprvisor has to evaluate the conditions given to determine a possible cause of the event and the correct procedure to use.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- Steps 2.0 through 8.0 may be performed in any order.
- Foldout CIP should be opened.

	STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____	**1.0**	**CHECK Plant Mode - IN MODE 1, 2, OR 3**	GO TO Step 4.0.
_____	**2.0**	**CHECK SG Levels - ALL LEVELS BEING CONTROLLED ON PROGRAM LEVEL IN AUTOMATIC**	IF using main feedwater to supply SGs, THEN PERFORM the following: **a.** PLACE any SG feedwater regulating valve(s) or low power feedwater regulating valve(s) **NOT** properly responding in MANUAL to match feedwater flow to steam flow: • SG 1A(2A) “NORM FCV-0551” • SG 1B(2B) “NORM FCV-0552” • SG 1C(2C) “NORM FCV-0553” • SG 1D(2D) “NORM FCV-0554” • SG 1A(2A) “LOW PWR FV-7151” • SG 1B(2B) “LOW PWR FV-7152” • SG 1C(2C) “LOW PWR FV-7153” • SG 1D(2D) “LOW PWR FV-7154” **b.** ADJUST affected SG feedwater regulating valve(s) as necessary to restore affected SG NR level(s) to between 68% and 74%. **c.** IF Feedwater Flow Transmitter FT-0510, FT-0520, FT-0530 or FT-0540 indicates channel failure, THEN PERFORM the following: **1)** ENSURE DA Storage Tank Level Control LK-7406 in MANUAL. **2)** *MAINTAIN* DA Storage Tank level between 65% and 80.

Step 2.0 continued on next page

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 2.0 continued from previous page

d. ENSURE an operable channel selected for affected feedwater regulating valve(s):

- feed flow
- steam flow
- SG level

e. PLACE affected SG feedwater regulating valve(s) in AUTO AND VERIFY proper operation.

f. IF SG NR level can **NOT** be maintained between 20% and 87.5%, THEN PERFORM the following:

1) TRIP the Reactor.

2) PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection.

_____ **3.0** **CHECK SGFP Master Speed Controller - RESPONDING PROPERLY IN AUTOMATIC**

IF using main feedwater to supply SGs, THEN PERFORM the following:

a. PLACE SGFP master speed controller in MANUAL AND ADJUST as necessary to maintain Feedwater/Steam DP per Addendum 1.

b. IF additional feed flow is necessary, THEN:

1) ENSURE SU SGFP is running.

2) START a Standby FW Booster Pump.

c. IF SG NR level can **NOT** be maintained between 20% and 87.5%, THEN PERFORM the following:

1) TRIP the Reactor.

2) PERFORM 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Exam Bank No.: 2159 **Last used on an NRC exam:** Never

SRO Sequence Number: 82

Unit 1 is operating at 100% power when the “CCW SURGE TK LVL LO” annunciator alarms.

Given the following conditions:

- The Crew has entered 0POP04-CC-0001, Component Cooling Water System Leak, due to the lowering Component Cooling Water Surge Tank level.
- Plant Operators in the field have NOT identified the source of the leak.
- Component Cooling Water Non-Vital Supply Valves closed as required at 64.6% level.
- Component Cooling Water Surge Tank continued to lower to 60% and is now stable.

Which of the following (1) identifies a possible leak location and (2) the correct action the Unit Supervisor should take based on the EXISTING plant conditions?

A. (1) A leak in the Letdown Heat Exchanger.
(2) Isolate letdown and enter 0POP04-CV-0004, Loss of Normal Letdown.

B. (1) A leak in RCP C Motor Air Cooler.
(2) Trip the Reactor, Turbine, and Reactor Coolant Pumps, then enter 0POP05-EO-EO00, Reactor Trip or Safety Injection.

C. (1) A leak in RCP C Motor Air Cooler.
(2) Enter 0POP04-TM-0005, Rapid Load Reduction and reduce power to less than P8 (40%) to allow tripping RCP C

D. (1) A leak in the Letdown Heat Exchanger.
(2) Isolate letdown and enter 0POP04-RC-0008, Boron Dilution Event.

Answer: B (1) A leak in RCP C Motor Air Cooler.
(2) Trip the Reactor, Turbine, and Reactor Coolant Pumps, then enter 0POP05-EO-EO00,
Reactor Trip or Safety Injection.

Exam Bank No.: 2159

K/A Catalog Number: APE 026 AA2.01 **Tier:** 1 **Group/Category:** 1

SRO Importance: 3.5 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to determine and interpret the following as they apply to the Loss of Component Cooling Water: Location of leak in the CCWS

STP Lesson: LOT 500.01 **Objective Number:** 92108

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

Reference: 0POP04-CC-0001, Loss of Component Cooling Water, pages 4, 9, and 10

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Leak location is credible because CCW supplies cooling to the letdown heat exchanger, but not correct since the surge tank level continued to lower after the first isolation. Action is credible since a loss of cooling to the heat exchanger would necessitate removing letdown from service.

B: CORRECT: A leak on an RCP would still be occurring after the first CCW isolation (64.6%), but stop after the second isolation (61.5%). The second isolation will isolate CCW to all RCPs so manual trip would be required.

C: INCORRECT: Action is credible because P8 allows for three RCP operation less than 40% power (although plant procedures do not). Also, action is not correct because plant conditions affects all RCPs and not just RCP C.

D: INCORRECT: Leak location is credible because CCW supplies cooling to the letdown heat exchanger, but not correct since the surge tank level continued to lower after the first isolation. Action is credible because a tube leak in other CVCS heat exchangers (i.e. seal water heat exchanger) would cause a dilution of the RCS, however this procedure would not be used in Mode 1

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate the given conditions and then determine which procedure would be appropriate to perform.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 2.0	*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 8.0.
____	**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.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 10.0	*MONITOR* **For CCW Surge Tank Low Level Commom 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 24.0. ---

Step 10.0 continued on next page

Step 10.0 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"

_____ **11.0 CHECK Plant MODE – MODE 1 or 2**

ENSURE RCPs secured.

_____ **a. TRIP the Reactor**

_____ **b. SECURE RCPs**

_____ **c. PERFORM 0POP05-EO-EO00, Reactor Trip Or Safety Injection**

_____ **d. CONTINUE Actions Of This Procedure As Resources Permit**

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.

_____ **12.0 CHECK Charging System Status - ANY PUMP RUNNING**

GO TO Step 17.0.

_____ **13.0 CHECK Centrifugal Charging Pump 1A(2A) Status - RUNNING**

GO TO Step 15.0.

Exam Bank No.: 2218 **Last used on an NRC exam:** Never

SRO Sequence Number: 83

Given the following:

- Unit 1 is in Mode 5
- RCS Boron Concentration is currently at 2810 ppm
- Control Rods are locked in the refueling position
- Preparations are being made for refueling operations
- Reactor Trip breakers are closed for maintenance and testing
- Rod Drive MG Sets are running for maintenance and testing

Subsequently:

- A fire breaks out in the Train C ESF Switchgear room.
- The Fire Brigade Leader reports a concern that the fire appears to be spreading toward the Rod Drive MG Set Room which could threaten the ability to move the Control Rods.

Based on the given information, which of the following describes the actions the Unit Supervisor should perform?

A. Direct I&C to restore Control Rod operation per 0PMP07-DM-0003, Rapid Refueling Rod Holdout Operation, and then ensure all Control Rods are fully inserted.

B. Enter 0POP04-RS-0001, Control Rod Malfunction, open the Reactor Trip Breakers and then ensure all Control Rods are fully inserted.

C. Direct a Plant Operator to secure the Rod Drive MG Sets per 0PMP07-DM-0003, Rapid Refueling Rod Holdout Operation, and then ensure all Control Rods are fully inserted.

D. Enter 0POP04-RS-0001, Control Rod Malfunction, place all Control Rod Lift Coil Disconnect Switches in the DISCONNECT position and then ensure all Control Rods are fully inserted.

Answer: A Direct I&C to restore Control Rod operation per 0PMP07-DM-0003, Rapid Refueling Rod Holdout Operation, and then ensure all Control Rods are fully inserted.

Exam Bank No.: 2218

K/A Catalog Number: APE 067 AA2.16 **Tier:** 1 **Group/Category:** 2

SRO Importance: 4.0 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to determine and interpret the following as they apply to the Plant Fire on Site:
Vital equipment and control systems to be maintained and operated during a fire.

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: LOT 505.01 Instruction on 0POP04-ZO-0008, Fire/Explosion

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: 0PMP07-DM-0003 is the only procedure used that will lock or unlock the control rods for rapid refueling. Restoring the operation of the control rods from a locked out rapid refueling position, would allow the Unit Suprvisor to direct the control rods be fully inserted or dropped into the reactor core.

B: INCORRECT: Credible because POP04-RS-0001 directs recovery from control rod malfunctions. Under normal circumstances, opening the trip breakers would cause control rods to fall, but not with the given conditions.

C: INCORRECT: Credible because 0PMP07-DM-0003 will provide instructions on how to make rods capable of insertion under these conditions, however securing the MG sets is not one of them and will not work with the conditions given.

D: INCORRECT: Credible because POP04-RS-0001 directs recovery from control rod malfunctions and provides instruction for operation of the lift coil disconnect switches, but for other reasons (recovery of a dropped rod).

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate plant conditions and then select the appropriate procedure. Also, there must be a knowledge of how the Control Rods operate normally and when placed in the Rapid Refueling position.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 16.0	**PERFORM 0POP04-ZO-0004, Personnel Emergencies, Concurrently With This Procedure**	
____ 17.0	***MONITOR* With Fire Brigade Leader For Smoke Affecting Adjacent Buildings**	GO TO Step 19.0.
____ 18.0	**DETERMINE If Any Of The Following Actions Are Required For Smoke Affecting Adjacent Buildings:** • Evacuate affected building(s) • SECURE OR REALIGN HVAC in affected building(s) • VENTILATE affected building(s)	
____ 19.0	**CHECK The Following Conditions exist:** • Control Rods are locked in the rapid refueling position • The Reactor Vessel head is – ON • The fire may degrade the ability to move the control rods	GO TO Step 21.0.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___ 20.0	**PERFORM The Following:** ___ a. **DIRECT I&C to restore control rod operation per 0PMP07-DM-0003, Rapid Refueling Rod Holdout Operation** ___ b. **Fully INSERT all Control Rods**	
___ 21.0	**CHECK The Fire Alarm Indicates A Possible RCB Cable Tray Fire**	GO TO Step 23.0.
___ 22.0	**PERFORM, Addendum 1, Response To RCB Cable Tray Alarms**	
___ 23.0	***MONITOR* A Valid Hi Temperature Alarm Received For Any Of The Following Filter Units:** • Containment Carbon Units • FHB Exhaust Filter Units • EAB Filter Units • MAB Filter Units	CONTINUE at Step 25.0 while monitoring filter unit temperature alarms AND PERFORM Step 24.0 when any valid filter unit temperature alarm is received.

<u>Exam Bank No.:</u> 2161 **Last used on an NRC exam:** Never

SRO Sequence Number: 84

Unit 1 is operating at 100% Power.

The crew has implemented 0POP04-RC-0004, Steam Generator Tube Leakage, due to the following current Radiation Monitor Readings given to the Unit Supervisor for the Steam Generators.

Steam Generators	A	B	C	D
Steam Line Radiation	1.8E-2 uCi/cc	1.5E-2 uCi/cc	1.4E-2 uCi/cc	3.9E-1 uCi/cc
Blowdown Radiation	3.1E-4 uCi/cc	2.4E-4 uCi/cc	2.3E-4 uCi/cc	4.6E-2 uCi/cc
N-16 Monitors	9.0 gpd	0.2 gpd	0.1 gpd	77.0 gpd

- Chemistry reports total current primary to secondary leak rate is 75 gpd.
- Leakage rate is rising 4 gpd/hr.

Which Steam Generator(s) have tube leaks and what action will the Unit Supervisor perform?

A. Only Steam Generator ‘D’ has a tube leak – Place the Unit in MODE 3 in less than or equal to 24 hours using 0POP03-ZG-0006, Plant Shutdown From 100%.

B. Only Steam Generator ‘D’ has a tube leak – Reduce power until the turbine is tripped using 0POP04-TM-0005, Fast Load Reduction.

C. Steam Generators ‘A’ and ‘D’ have tube leaks – Reduce power until the turbine is tripped using 0POP04-TM-0005, Fast Load Reduction.

D. Steam Generators ‘A’ and ‘D’ have tube leaks – Place the Unit in MODE 3 in less than or equal to 24 hours using 0POP03-ZG-0006, Plant Shutdown From 100%.

Answer: A Only Steam Generator ‘D’ has a tube leak; Place the Unit in MODE 3 in less than or equal to 24 hours using 0POP03-ZG-0006, Plant Shutdown From 100%.

Exam Bank No.: 2161

K/A Catalog Number: APE 037 AA2.02 **Tier:** 1 **Group/Category:** 2

SRO Importance: 3.9 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: Agreement/disagreement among redundant radiation monitors

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-0004, Steam Generator Tube Leak.

Attached Reference ☑ **Attachment:** 0POP04-RC-0004, Steam Generator Tube Leak, Page 5 (Step 8) and Page 34 (Add 6)

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: The elevated leak rate from the N-16 monitor for SG 'A' is due to it being close to SG 'D' steam line. The N-16 monitors are in the TGB where there is no concrete to shield in between the steam lines. With the Steam Line and Blowdown Rad monitors shielded in the IVC and not being elevated on all but SG 'D', the comparison would indicate that only SG 'D' had a tube leak. With one SG leaking greater than 75 gpd with a rate of change less than 30 gpd/hr, the correct action is to use 0POP03-ZG-0006 and be in MODE 3 in less than 24 hours.

B: INCORRECT: Credible because this action would be correct if the leakrate was escalating faster (>= 30 gpd/hr).

C: INCORRECT: Credible because SG A N-16 reading is considerable larger than B or C, but this is due to shine from the SG D steam line. Credible action because this action would be correct if the leakrate was escalating faster (>= 30 gpd/hr)

D: INCORRECT: Credible because SG A N-16 reading is considerable larger than B or C, but this is due to shine from the SG D steam line.The D steam line is on the outside 'A' is only one adjacent to 'D'.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate the given conditions and then select the appropriate procedure.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 6.0	***MAINTAIN* VCT Level – GREATER THAN 15% WITH CHARGING PUMP SUCTION ALIGNED TO VCT (CP004)** • Auto makeup • Manual makeup	PERFORM the following: **a.** TRIP the Reactor. **b.** INITIATE Safety Injection. **c.** GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.
____ 7.0	**CHECK Main Turbine In Service**	GO TO Step 13.0.
____ 8.0	**CHECK For One Of The Following:** • **Leakage From Any One SG Is GREATER THAN OR EQUAL TO 75 gpd <u>AND</u> Continues To Increase At GREATER THAN OR EQUAL TO 30 gpd/hr** **OR** • **Leakage From Any One SG Is GREATER THAN OR EQUAL TO 75 gpd <u>AND</u> Loss of Continuous Radiation Monitoring**	PERFORM the following: **a.** COMPARE SG Tube Leak Rates to values listed in Addendum 6. **b.** <u>IF</u> Mode 3 is REQUIRED by Addendum 6, <u>THEN</u> PERFORM the following: **1)** COMMENCE plant shut down per 0POP03-ZG-0006, Plant Shutdown From 100% To Hot Standby, per response time requirements of Addendum 6. **2)** GO TO Step 10.0. **c.** <u>IF</u> Shutdown <u>NOT</u> required, <u>THEN</u> RETURN to procedure and step in effect.
____ 9.0	**PERFORM The Following:** ____ **a. COMMENCE Plant Shutdown Per 0POP04-TM-0005, Fast Load Reduction Per The Response Time Requirements Of Addendum 6** ____ **b. CONTINUE performance of 0POP04-TM-0005, Fast Load Reduction until TURBINE TRIPPED**	

Addendum 6	Recommended Response Times	Addendum 6 Page 1 of 1

CAUTION

Post Reactor Shutdown conditions in the primary to secondary leakage (RCS temperature and pressure decreasing and SG pressure increasing) may reduce the SG Tube Leakage Rate. Plant Shutdown and Cooldown Rates should be based on the initial or increased leakage and **NOT** reduced leakage due to the Post Shutdown conditions.

Action Level	Leak Rate	Increasing Leak Rate	Response Times (1)
3	≥ 75 gpd (4)	Rate of inc ≥ 30 gpd/hr	Reduce Rx PWR to ≤ 50% in 1 hr AND Mode 3 in the next 2 hr
3	≥ 75 gpd AND Loss of Continuous Radiation Monitoring (3)	NA	Reduce Rx PWR to ≤ 50% in 1 hr AND Mode 3 in the next 2 hr
3	≥ 150 gpd	NA	Mode 3 ≤ 6 hr
2	≥ 75 gpd (4)	Rate of inc < 30 gpd/hr	Mode 3 ≤ 24 hr
1	< 75 gpd AND Loss of Continuous Radiation Monitoring (3)	N/A	Continued Operations (2)
1	≥ 30 gpd	N/A	Continued Operations (2)
Increased Monitoring	≥ 5 gpd	N/A	Continued Operations (2)

(1) Response times are the maximum times allowed. Power Reduction and Mode Change(s) may be completed in less time.

(2) Continued Operations per Plant Management direction. Refer to 0PGP03-ZO-0041, Action For Monitoring Primary to Secondary Leakage.

(3) Loss of Continuous Radiation Monitoring as defined in 0PGP03-ZO-0041, Action for Monitoring Primary to Secondary Leakage.

(4) With a continued increase in leakage rate over 30 minute time interval per the next column.

Exam Bank No.: 2162 **Last used on an NRC exam:** Never

SRO Sequence Number: 85

Unit 2 is operating at 60% Power.

Subsequently:

- A Reactor Operator reports that Condenser Vacuum is 23" HG and stable and Main Generator Megawatt output is slowly lowering.
- Further investigation reveals that the "MAIN COND VACUUM LO" alarm is EXTINGUISHED and the "C9 COND AVAILABLE FOR STEAM DUMP" light is ILLUMINATED.

Based on the given plant conditions, which of the following describes actions that the Unit Supervisor should perform?

A. (1) Trip the Reactor, Ensure the Main Turbine tripped and enter 0POP05-EO-EO00, Reactor Trip or Safety Injection.
(2) Write a Condition Report on the "MAIN COND VACUUM LO" alarm which is NOT responding properly.

B. (1) Trip the Reactor, Ensure the Main Turbine tripped and enter 0POP05-EO-EO00, Reactor Trip or Safety Injection.
(2) Write a Condition Report on the "MAIN COND VACUUM LO" alarm and the "C9 COND AVAILABLE FOR STEAM DUMP" light. Both are NOT responding properly.

C. (1) Enter 0POP04-CR-0001, Loss of Condenser Vacuum, and begin a Turbine Load Reduction.
(2) Write a Condition Report on the "MAIN COND VACUUM LO" alarm and the "C9 COND AVAILABLE FOR STEAM DUMP" light. Both are NOT responding properly.

D. (1) Enter 0POP04-CR-0001, Loss of Condenser Vacuum, and begin a Turbine Load Reduction.
(2) Write a Condition Report on the "MAIN COND VACUUM LO" alarm which is NOT responding properly.

Answer: D (1) Enter 0POP04-CR-0001, Loss of Condenser Vacuum, and begin a Turbine Load Reduction.
(2) Write a Condition Report on the 'MAIN COND VACUUM LO' alarm which is NOT responding properly.

Exam Bank No.: 2162

K/A Catalog Number: APE 051 G2.4.46 **Tier:** 1 **Group/Category:** 2

SRO Importance: 4.2 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Condenser Vacuum: Ability to verify that the alarms are consistent with the plant conditions.

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

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

Reference: 0POP04-CR-0001, Loss of Condenser Vacuum

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because conditions can exist where a manual trip is required, but not until 21 inches of vacuum.

B: INCORRECT: Credible because conditions can exist where a manual trip is required, but not until 21 inches of vacuum. Part 2 is credible because both the alarm and the light are dependent on vacuum, however the setpoint for the light is 22 inches.

C: INCORRECT: Part 2 is credible because both the alarm and the light are dependent on vacuum, however the setpoint for the light is 22 inches.

D: CORRECT: Condenser vacuum is above 21" HG but below 26" HG. A Turbine Load reduction is required. The "MAIN COND VACUUM LO" alarm did not annunciate at 26" HG when it was supposed to.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate the given conditions and then determine the correct action to take.

PURPOSE

This procedure provides guidelines for operator response to a lowering or reduced Main Condenser vacuum.

The procedure has two levels of response. The first response is a diagnostic process to identify and correct the cause of the loss of vacuum. The second is to monitor condenser vacuum and ensure automatic actuations occur at the required setpoint.

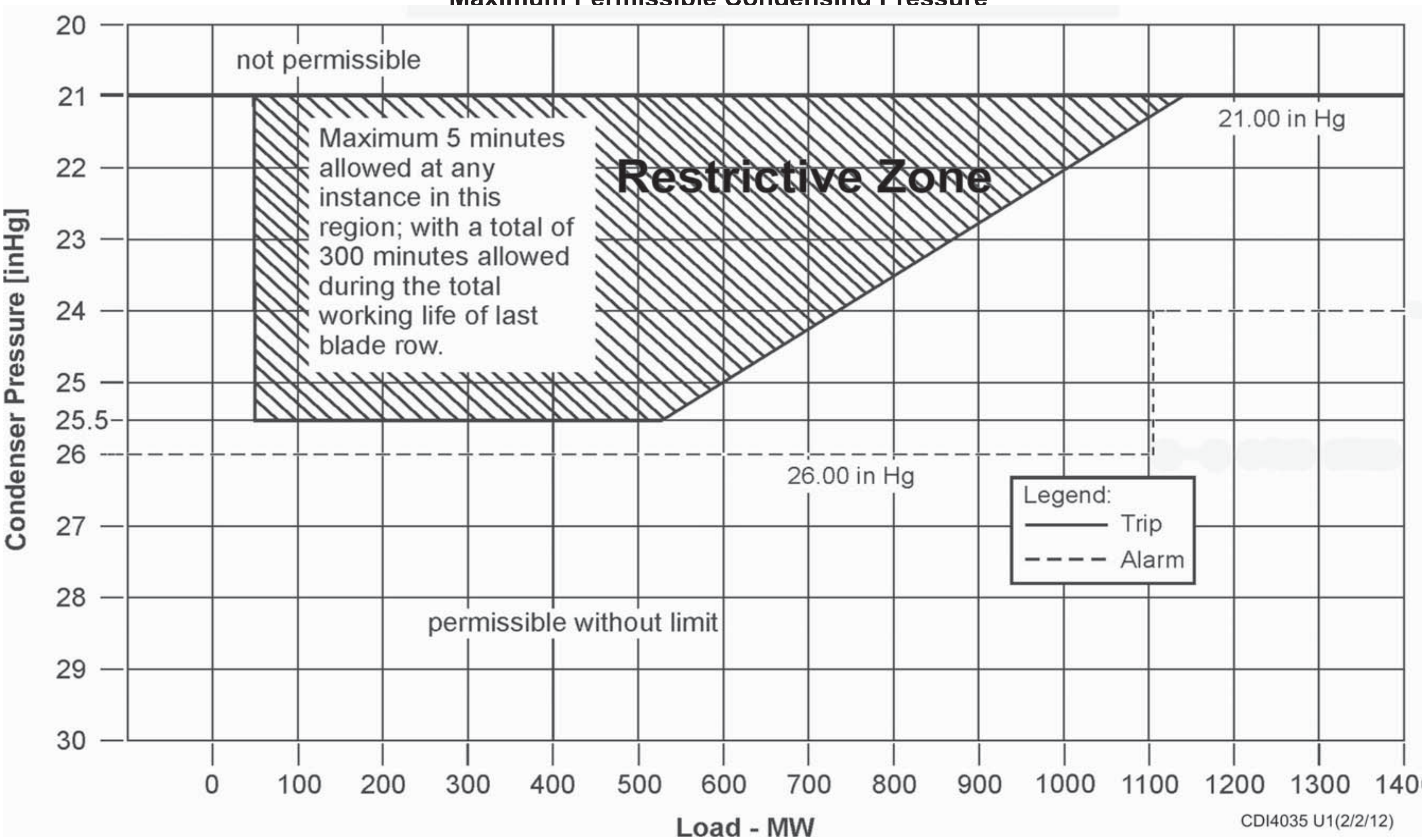
Condenser Vacuum (in. Hg)	Automatic Actuation
(UNIT 1 ONLY) > 80% Power 24	Condenser Vacuum Low alarm
(UNIT 1 ONLY) ≤ 80% Power 26	Condenser Vacuum Low alarm
(UNIT 2 ONLY) 26	Condenser Vacuum Low alarm
26	Standby Condenser Air Removal Pump starts
22	C-9, Steam Dump Block Permissive
21	Main Turbine Trip

SYMPTOMS OR ENTRY CONDITIONS

1. The following Control Room annunciator alarm:
   - "MAIN COND VACUUM LO" Lampbox 07M3 Window E-7
   - "CWP TRIP/FAIL START" Lampbox 09M1 Window A-3
2. Automatic starting of the standby Condenser Air Removal (CARs) pump.
3. Unexplained continuing lowering in Main Turbine Load.
4. Lowering Main Condenser vacuum.

Unit 1

Exhaust Pressure Limitations
Maximum Permissible Condensing Pressure



This Procedure is Applicable anytime a vacuum exists in the Main Condenser

Exam Bank No.: 2216 **Last used on an NRC exam:** Never

SRO Sequence Number: 86

Instrument air has been lost to CV-FV-0011.

Which of the following correctly describes the action to be taken by the control room staff and why?

The Unit Supervisor should enter …

A. 0POP04-RP-0002, Loss Of Automatic Pressurizer Level Control, and close FCV-0205 to minimize thermal stress on the charging nozzle at the RCS pipe.

B. 0POP04-CV-0004, Loss of Normal Letdown, and close FCV-0205 to minimize thermal stress on the charging nozzle at the RCS pipe.

C. 0POP04-RP-0002, Loss Of Automatic Pressurizer Level Control, and close FCV-0205 to maintain VCT level and prevent a loss of suction to the charging pump.

D. 0POP04-CV-0004, Loss of Normal Letdown, and close FCV-0205 to maintain VCT level and prevent a loss of suction to the charging pump.

Answer: B 0POP04-CV-0004, Loss of Normal Letdown, and close FCV-0205 to minimize thermal stress on the charging nozzle at the RCS pipe.

Exam Bank No.: 2216

K/A Catalog Number: APE 065 G2.1.32 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.0 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Instrument Air: Ability to explain and apply system limits and precautions.

STP Lesson: LOT 201.06 **Objective Number:** 48669

In regard to POP02-CV-0004, POP02-CV-0005, AND PSP03-CV-0011, DESCRIBE the following: 1. Purpose and Scope, 2. Precautions, and 3. Notes and Cautions

Reference: POP04-CV-0004, step 2.0 basis, POP02-CV-0004, step 4.14

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Procedure is credible because pressurizer level will rise during this event as it will during a low failure of the controlling level channel which POP04-RP-0002 is designed to address; therefore it would not be a correct entry in this instance.

B: CORRECT: Loss of IA to the valve will cause it to close and result in letdown flow going to 0. POP04-CV-0004 will address this condition. If letdown flow is lost, then charging should be isolated to minimize thermal stress on the charging nozzle since preheating is no longer occurring.

C: INCORRECT: Procedure is credible because pressurizer level will rise during this event as it will during a low failure of the controlling level channel which POP04-RP-0002 is designed to address; therefore it would not be a correct entry in this instance. Reason is credible because this failure will cause VCT level to lower and closing FCV-0205 will minimize that, however loss of suction to the charging pump is not a concern due to auto makeup and low level swapover to the RWST.

D: INCORRECT: Reason is credible because this failure will cause VCT level to lower and closing FCV-0205 will minimize that, however loss of suction to the charging pump is not a concern due to auto makeup and low level swapover to the RWST.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must determine the effect the given failure will have and select the proper procedure. A knowledge of system design/limits is needed to deterrmine the reason for the action taken.

PURPOSE

This procedure provides guidance for Operator response due to a loss of Normal Letdown.

This procedure provides guidance for recovery actions including response due to an instrument failure, letdown line failure or a High Energy Line Break (HELB). The following is a list of valves in the letdown path and possible causes of closure.

	Pzr Level < 17%	Containment Isol Phase A	High CVCS Room Temp	Loss of Power/Air	Loss of signal from Letdown Pressure PT-0135
LCV-0465	√				
LCV-0468	√				
FV-0011	√	√		√	
FV-0012				√	
FV-0013				√	
MOV-0014					
MOV-0023		√	√		
MOV-0024		√	√		
PCV-0135					√

SYMPTOMS OR ENTRY CONDITIONS

1. Any of the following Control Room annunciator alarms:
   - "LETDN HX OUTL PRESS HI" Lampbox 4M08, Window C-4
   - "LETDN HX OUTL FLOW HI/LO" Lampbox 4M08, Window D-4
2. The following symptoms are indicative of a loss of Normal Letdown
   - Low Letdown flow as indicated on FI-0132
   - HELB Actuation as indicated by closure of MOV-0023 or MOV-0024, letdown containment isolation valves
3. This procedure is entered any time Normal Letdown is isolated unexpectedly. This procedure is NOT entered if loss of letdown is due to a Safety Injection.

STEP DESCRIPTION FOR 0POP04-CV-0004 STEP 2.0

STEP: VERIFY Letdown Containment Isolation Valves - OPEN

- ICIV MOV-0023
- OCIV MOV-0024

PURPOSE: To determine if either of these valves have closed and take additional action if either valve has closed.

BASIS: These valves close to isolate a high energy line break in the MAB resulting from letdown line breaks. If either of these valves have closed additional upstream valves must be closed manually to prevent a loss of coolant through PSV-3100, which is between the letdown orifices and the containment isolation valves. (Ref. P&ID 9F05005)

ACTIONS: If either valve has closed, manually close FV-0011, which ensures that letdown flow is isolated. The charging flow control valve (FCV-0205) is closed, the miniflow recirc valve is opened for the operating charging pump and RCP seal injection valve (HCV-0218) is adjusted to maintain RCP seal injection within the specified band. The orifice valves are closed, followed by LCV-0465 and 0468. It is necessary to close FV-0011 following the closure of MOV-0023 and MOV-0024 to minimize the amount of time the letdown relief valve is challenged.

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: Switches for FV-0011, MOV-0014, FV-0013, FV-0012, LCV-465, LCV-468, FCV-0205, HCV-0218 on CP004.

KNOWLEDGE: Part of isolating letdown is also to isolate charging to minimize thermal shock at the charging nozzle at the RCS pipe (Ref. PLS 5Z010ZS1101, page 47). Before normal letdown is restored, an evaluation should be made to verify it is feasible, i.e., any leakage points isolated do not prevent letdown from being restored.

STEP DESCRIPTION FOR 0POP04-CV-0004 STEP 6.0

STEP: VERIFY Letdown Orifice Header Isolation Valve “LETDN ORIF HDR ISOL FV-0011” - OPEN

PURPOSE: Determine if a malfunction has occurred with valve FV-0011.

BASIS: Valve FV-0011 closes on low pressurizer level or a Containment Phase A isolation signal (SSPS Train C). Both of these conditions have been checked in previous steps. The other possible causes for FV-0011 to close is a loss of instrument air or loss of electrical power. FV-0011 may also close on a temporary loss of power to LT-0465 or LT-0468, which may not close LCV-0465 or LCV-0468. This step determines if FV-0011 is open or closed and investigates for possible causes if the valve has closed.

ACTIONS: The determination is made if FV-0011 is open or closed. If the valve is closed instrument air is checked and the power supply to the valve is checked. If the source of the problem is identified and corrected, the letdown orifices are closed, and then letdown can be restored per Addendum 4, Placing Normal Letdown In Service.

INSTRUMENTATION: N/A

CONTROL/EQUIPMENT: Switches for MOV-0014, FV-0013, FV-0012, PK-0135, FV-0011 on CP004.

KNOWLEDGE: FV-0011 fails closed on loss of air or loss of power to the solenoid.

Exam Bank No.: 2182 **Last used on an NRC exam:** Never

SRO Sequence Number: 87

Unit 1 was at 100% power when an event occurred that required a fast load reduction to 80% power. During the power reduction, 2 rods in Control Bank 'D' failed to move while the remainder of the rods in the bank inserted 18 steps.

Which of the following describes the ACTION the Unit Supervisor should now take and the EFFECTS this condition may have on Unit 1?

	ACTION	EFFECTS
A.	Attempt to re-align the Control Rods per 0POP04-RS-0001, Control Rod Malfunction, Addendum 2, Recovery of Misaligned Rods.	Fuel Integrity may be challenged by Power Peaking Factor limits being exceeded.
B.	Place Unit 1 in MODE 3 within 6 hours per 0POP03-ZG-0006, Plant Shutdown from 100%.	Fuel Integrity may be challenged by DNBR limits being exceeded.
C.	Attempt to re-align the Control Rods per 0POP04-RS-0001, Control Rod Malfunction, Addendum 2, Recovery of Misaligned Rods.	Fuel Integrity may be challenged by DNBR limits being exceeded.
D.	Place Unit 1 in MODE 3 within 6 hours per 0POP03-ZG-0006, Plant Shutdown from 100%.	Fuel Integrity may be challenged by Power Peaking Factor limits being exceeded.

Answer: D Place Unit 1 in MODE 3 within 6 hours per 0POP03-ZG-0006, Plant Shutdown from 100%; Fuel Integrity may be challenged by Power Peaking Factor limits being exceeded.

Exam Bank No.: 2182

K/A Catalog Number: APE 005 G2.1.43 **Tier:** 1 **Group/Category:** 2

SRO Importance: 4.6 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Inoperable/Stuck Control Rod: Ability to use procedures to determine the effects on reactivity of plant changes, such as reactor coolant temperature, secondary plant, fuel depletion, etc.

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, Control Rod Malfunction and TS 3.1.3.1 Action d.

Attached Reference ☑ **Attachment:** 0POP04-RS-0001, pages 1-10

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Action is credible because the procedure contains actions to align rods, but only if there is only one rod affected.

B: INCORRECT: Effect is credible because DNBR limits are often a concern when power is affected, however the limits are not expected to be exceeded with misaligned control rods.

C: INCORRECT: Action is credible because the procedure contains actions to align rods, but only if there is only one rod affected. Effect is credible because DNBR limits are often a concern when power is affected, however the limits are not expected to be exceeded with misaligned control rods.

D: CORRECT: With 2 rods misaligned a plant shutdown is required. Fuel Integrity is challenged by Power Peaking Factors possibly being exceeded.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor must evaluate the given condition and choose the correct procedure and identify the effect on the Unit caused by the condition.

_____ **4.0 CHECK For Misaligned Rods:**

Step	Actions/Expected Response	Response Not Obtained
_____	**a. CHECK All Rods – ANY RODS MISALIGNED**	**a.** GO TO Step 5.0.
_____	**b. CHECK All Rods – ONLY ONE ROD MISALIGNED**	**b.** IF two or more rods are **NOT** aligned in MODES 1 OR 2, THEN: **1)** REFER TO Technical Specification 3.1.3.1 Action d for appropriate action. **2)** 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. **3)** 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.

Exam Bank No.: 2211 **Last used on an NRC exam:** Never

SRO Sequence Number: 88

A large leak of radioactive water is occurring in the Mechanical Auxiliary Building.

The following sumps are experiencing a rise in level due to the leak.

1. Component Cooling Water Sump
2. Mechanical Auxiliary Building Floor Drain Sump #2
3. Essential Cooling Water Sump
4. Mechanical Auxiliary Building Elevator Sump

Under these conditions securing the sump pumps for which of the above sumps would prevent an unmonitored release directly to the environment?

A. 1

B. 2

C. 3

D. 4

Answer: C 3

Exam Bank No.: 2211

K/A Catalog Number: G2.3.11 **Tier:** 3 **Group/Category:** 3

SRO Importance: 4.3 **10CFR Reference or SRO Objective:** 55.43(b)(4)

Ability to control radiation releases.

STP Lesson: LOT 203.10 **Objective Number:** 98076

Given a set of conditions, PREDICT the effect(s) and/or response(s) on the Equipment and Floor Drains system.

Reference: LOT 203.10 lesson on Equipment and Floor Drains

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: This is a credible distractor because Component Cooling Water Sump could collect radioactive effluent but the sump is pumped to the Condensate Polisher Regeneration Waste Collection Tank (CPRWCT) and then to a WHT where the water is processed and monitored prior to release.

B: INCORRECT: This is a credible distractor because MAB Sump #2 could collect radioactive effluent but is pumped to the Floor Drain Tank and then to a WHT where the water is processed and monitored prior to release.

C: CORRECT: Essential Cooling Water Sump pumps directly to the enviroment through the Circ Water system without being monitored.

D: INCORRECT: This is a credible distractor because MAB Elevator Sump could collect radioactive effluent but is pumped to MAB Sump #3, then to the Floor Drain Tank and then to a WHT where the water is processed and monitored prior to release.

Question Level: F **Question Difficulty** 3

Justification:

In order to control a potential release, the Unit Supervisor needs fundemental knowledge of where MAB sumps discharge.

ESSENTIAL COOLING WATER SUMP

- Location - 10’MAB behind Essential Chillers
- Not potentially contaminated, reduces amount to be processed
- Raised ledge around ECW Sump and Drain Hubs
- Input - various non-radioactive drains in MEAB (CCW Pump Area drains, Essential Chillers ECW Relief's, EAB HVAC Sump)
- Output - pumped to Circ Water Discharge piping

Exam Bank No.: 2180 **Last used on an NRC exam:** Never

SRO Sequence Number: 89

Given the following:

- A Waste Monitor Tank release needs to be performed
- RT-8038, LWPS Monitor #1, was declared inoperable 3 days ago

In accordance with 0PSP07-WL-LDP2, Liquid Effluent Permit with RT-8038 Inoperable, who approves the discharge and what are the additional requirements for the release per the Offsite Dose Calculation Manual (ODCM)?

	Approval	ODCM requirement
A.	Shift Manager	Continuous surveys of the discharge piping during the release
B.	Shift Manager	At least two independent samples of the monitor tank are analyzed prior to the release
C.	Health Physics Manager	Continuous surveys of the discharge piping during the release
D.	Health Physics Manager	At least two independent samples of the monitor tank are analyzed prior to the release

Answer: B Shift Manager; At least two independent samples of the monitor tank are analyzed prior to the release

Exam Bank No.: 2180

K/A Catalog Number: G2.3.6 **Tier:** 3 **Group/Category:** 3

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(2)

Ability to approve release permits.

STP Lesson: **Objective Number:** SRO-13400

AUTHORIZE a release of liquid waste in accordance with 0PSP07-WL-LDP1, Liquid Effluent Permit.

Reference: 0PSP07-WL-LDP2 step 5.1.23, ODCM Table 3.3-12

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because continuous surveys could give an indication of a release of excess activity, however it does not match the requirments

B: CORRECT: Approval and requirements correct per the given references

C: INCORRECT: Credible because Health Physics is involved with all things concerned with radiation, but they are not the approval authority for a discharge. Although continuous surveys could give an indication of a release of excess activity, it does not match the requirments

D: INCORRECT: Credible because Health Physics is involved with all things concerned with radiation, but they are not the approval authority for a discharge.

Question Level: F **Question Difficulty** 3

Justification:

The SRO applicant must have a knowledge of release and ODCM requirements

Initials

5.1.20.5 Document jumper installation in the Effluent Release Logbook. _____

5.1.21 IF the discharge will occur on the fourteenth day **OR** greater from the time the monitor was declared inoperable, AND the discharge will proceed, THEN perform the following:

5.1.21.1 Generate a Condition Report to document making releases with RT-8038 inoperable ≥ 14 days. (Or record CR# if a CR was already generated on a previous release) _____

CR #_____________________

5.1.21.2 Obtain Chemistry Manager or Designee approval for release(s) to occur with RT-8038 inoperable ≥ 14 days: _____

Approval obtained (Initials):_________Date: ________ Time: ________

5.1.22 Perform an Independent Verification of the Liquid Effluent Permit.

Independent Verification by: (Init.)__________ Date:_________ Time:________

5.1.23 Forward package to Shift Manager for signature for the following: _____

- Review of the Liquid Effluent Permit
- Approval for the discharge

Shift Manager ____

Date: ______________ Time: ______________

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1.	Radioactivity Monitors Providing Alarm and Automatic Termination of Release		
	Liquid Waste Processing Discharge Monitor (N1RA-RT-8038 or N2RA-RT-8038)	1	43
2.	Flow Rate Measurement Devices		
	Liquid Waste Processing Discharge Line (N1WL-FT-4078 or N2WL-FT-4078)	1	46

ACTION STATEMENTS

ACTION 43 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 14 days provided that prior to initiating a release:

a. At least two independent samples are analyzed in accordance with Control 4.11.1.1.1, and

b. At least two technically qualified members of the facility staff independently verify the release rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 44 - (Not used)

ACTION 45 - (Not used)

ACTION 46 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

Exam Bank No.: 2217 **Last used on an NRC exam:** Never

SRO Sequence Number: 90

Given the following:

- Unit 1 is at 100% power.
- Train ‘A’ ECW Pump is inoperable to allow for maintenance.

Subsequently:

- Electrical Maintenance requests approval to perform Troubleshooting on Train ‘B’ ECW Pump Room Fan ‘11B’ breaker which showed a hot spot during a thermal imaging scan 2 weeks ago.

In accordance with 0POP01-ZO-0012, Operations Troubleshooting Process, the ____________ Manager would approve troubleshooting activities. In the given scenario Troubleshooting activities ____________ be allowed.

A. Shift, would NOT

B. Shift, would

C. Maintenance, would

D. Maintenance, would NOT

Answer: A Shift, would NOT

Exam Bank No.: 2217

K/A Catalog Number: G2.2.20 **Tier:** 3 **Group/Category:** 2

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(3)

Knowledge of process for managing troubleshooting activities.

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-ZO-0012, Operations Troubleshooting Process, steps 5.1 and 6.1.13

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Modified **Modified From** 2134

Distractor Justification

A: CORRECT: Per POP01-ZO-0012, the shift manager would approve troubleshooting activities. In this case, since it is a cross train related component which could become inoperable, troubleshooting would not be allowed.

B: INCORRECT: Credible because the procedure does allow for cross train troubleshooting provided the system will not be rendered inoperable. In this case, if the vent fan becomes inoperable, the system is inoperable.

C: INCORRECT: Credible because the Maintenance manager must approve certain work activities (i.e. work behind a freeze seal, work on main/backseated valves) and electrical maintenance is involved. The work allowance is credible because the procedure does allow for cross train troubleshooting provided the system will not be rendered inoperable.

D: INCORRECT: Credible because the Maintenance manager must approve certain work activities (i.e. work behind a freeze seal, work on main/backseated valves) and electrical maintenance is involved.

Question Level: F **Question Difficulty** 3

Justification:

The applicant must have knowledge of the Operations Troubleshooting process.

3.1.8 0POP01-ZO-0011, Operability, Functionality, and Reportability Guidance

3.2 Conduct of Operations Manual

3.3 CR 09-19595, Process Shortfall Concerning Operations Troubleshooting Activities

3.4 NRC NOV 95-06-02 (CR 95-6790), Operators Changed Configuration of Safety-Related Equipment Without Written Instructions

4.0 Prerequisites

4.1 OBTAIN Shift Manager approval for entry into this procedure. _____

5.0 Precautions and Notes

5.1 Troubleshooting should only be performed on operable systems which will not be rendered inoperable as determined by the Shift Supervisor, or on systems which have already been declared inoperable.

5.2 During Plant Operations troubleshooting activities, the Shift Manager SHALL maintain communications with the Division Manager and the Operations Manager as appropriate to keep them updated with progress and plant impact.

5.3 During performance of this procedure, a system, structure or component (SSC) may be found to be functioning other than as expected. 0POP01-ZO-0011 (Operability, Functionality, and Reportability Guidance) provides departmental guidance for determining Operability and Functionality status of SSCs that come into question.

5.4 Activities during troubleshooting SHALL NOT be performed using uncontrolled written instructions.

5.5 Whenever component manipulations are required during the troubleshooting process, they should be performed using approved plant procedures or CROEs.

5.6 The use of approved plant procedures or CROEs to assist in troubleshooting SHALL NOT warrant exit from this procedure. This procedure SHALL remain in effect throughout the entire troubleshooting process.

5.7 Once approved, any revisions to the original Troubleshooting Plan as documented in Addendum 2 SHALL require an additional LCR and additional approval signatures, including the Shift Manager.

5.8 Conditions/problems discovered during the troubleshooting process SHALL be reported in accordance with 0PGP03-ZX-0002 (Condition Reporting Process).

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Exam Bank No.: 2134 Last used on an NRC exam: 2011

Given the following:

- Unit 1 is at 100% power.
- Train ‘A’ ECW Pump is inoperable to allow for maintenance.

Subsequently:

- During maintenance on Train ‘A’ ECW Pump, an unexpected condition arose on Train ‘A’ ECW Discharge Motor Operated Valve (MOV) and the craft wants to troubleshoot the MOV, however troubleshooting the MOV will render the MOV inoperable.

In accordance with 0POP01-ZO-0012, Operations Troubleshooting Process, troubleshooting activities on the MOV _________ allowed. Trouble shooting plans are approved by the ____________ Manager.

A. ARE, Shift

B. are NOT, Shift

C. are NOT, Plant

D. ARE, Plant

Answer: A ARE, Shift

Exam Bank No.: 2134 **RO Outline Number:** **SRO Outline Number:**

K/A Catalog Number: G2.2.20 **Tier:** 3 **Group/Category:** 2

RO/SRO Importance: 2.6 / 3.8 **RO-10CFR55.41** #

SRO-10CFR55.43 # 3 or **SRO Obj:**

Knowledge of process for managing troubleshooting activities.

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-ZO-0012, Operations Troubleshooting Process, steps 5.1 and 6.1.13

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from:**

Distractor Justification

A: CORRECT: Troubleshooting activities are allowed since the pump (system) are already inoperable. The Shift Manager is the correct approval authority.

B: INCORRECT: Troubleshooting activities would not be allowed if the train was operable, but are since the pump (system) are already inoperable. The approval authority is correct.

C: INCORRECT: Troubleshooting activities would not be allowed if the train was operable, but are since the pump (system) are already inoperable. The Plant Manager is the approval authority for many activities, however it is the Shift Manager in this case.

D: INCORRECT: Activity analysis is correct. The Plant Manager is the approval authority for many activities, however it is the Shift Manager in this case.

Level F **Difficulty** 3

Justification:

The applicant must have knowledge of the Operations troubleshooting process.

NOTE

The reviewer for technical adequacy SHALL NOT be the plan developer.

6.1.9 OBTAIN a review of the troubleshooting plan for technical adequacy from a Senior Reactor Operator, along with reviewer signature on Addendum 2. _____

6.1.10 ENSURE that a License Compliance Review (LCR) of the troubleshooting plan has been performed per 0PAP01-ZA-0103, License Compliance Review. _____

6.1.10.1 On the LCR form, RECORD this procedure number and revision. _____

6.1.10.2 In the "Procedure Title or Description" line of the LCR form, INCLUDE a brief description of the troubleshooting task to be performed. _____

6.1.10.3 ATTACH the LCR to Addendum 2 of this procedure. _____

6.1.10.4 ENSURE that the requirements of 0PGP03-ZA-0010 Section 6.8 have been met. _____

6.1.11 ENSURE that a Security Compliance Review (SCR) has been performed in accordance with 0PAP01-ZA-7358 Security Compliance Review. _____

6.1.11.1 ATTACH the SCR to Addendum 2 of this procedure. _____

6.1.12 ENSURE that a Work Risk Assessment has been performed per 0PGP03-ZA-0090, Work Process Program, and documented in the CR referred to in Step 6.1.2. _____

6.1.13 OBTAIN approval signature for the plan from the Shift Manager on Addendum 2. _____

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Exam Bank No.: 2163 **Last used on an NRC exam:** Never

SRO Sequence Number: 91

Unit 1 is at 100% Power.

A Reactor Operator reports the following to the Unit Supervisor:

- VCT level is 60% and slowly rising.
- A VCT Auto Make-up is NOT in service.
- Tavg is 1ºF below Tref and slowly lowering.
- "VCT HI/LO" alarm is illuminated.
- "VCT LO/LO" alarm is illuminated.

Which of the following (1) describes the cause of these indications and (2) the action the Unit Supervisor should take?

A. (1) VCT level transmitter, CV-LT-0113, has failed low causing an inadvertent boration.
(2) Enter 0POP09-AN-04M8, Annunciator Response for the VCT LO/LO alarm, OPEN CV-MOV-0113A, "VCT OUTL ISOL", and then de-energize the MOV.

B. (1) VCT level transmitter, CV-LT-0113, has failed low causing an inadvertent boration.
(2) Enter 0POP04-TM-0001, Turbine Load Rejection, and lower Main Turbine load to match Tref with the lower Tavg.

C. (1) VCT level transmitter, CV-LT-0112, has failed low causing an inadvertent boration.
(2) Enter 0POP09-AN-04M8, Annunciator Response for the VCT LO/LO alarm, OPEN CV-MOV-0112B, "VCT OUTL ISOL", and then de-energize the MOV.

D. (1) VCT level transmitter, CV-LT-0112, has failed low causing an inadvertent boration.
(2) Enter 0POP04-TM-0001, Turbine Load Rejection, and lower Main Turbine load to match Tref with the lower Tavg.

Answer: A (1) VCT level transmitter, CV-LT-0113, has failed low causing an inadvertent boration.
(2) Enter 0POP09-AN-04M8, Annunciator Response for the VCT LO/LO alarm, OPEN
CV-MOV-0113A, "VCT OUTL ISOL", and then de-energize the MOV.

Exam Bank No.: 2163

K/A Catalog Number: 004 A2.10 **Tier:** 2 **Group/Category:** 1

SRO Importance: 4.2 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertant boration/dilution

STP Lesson: LOT 201.06 **Objective Number:** 507226

Given a description of plant conditions, ANALYZE the conditions and PREDICT how the CVCS will respond.

Reference: LOT 201.06 Lesson Plan for CVCS

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: A low failure of CV-LT-0113 will close CV-MOV-0113A, VCT OUTL ISOL, and open CV-MOV-0113B, RWST TO CHG PUMP SUCTION ISOL effectively borating the RCS. The annunciator response for this failure will give direction to open and de-energize CV-MOV-0113A.

B: INCORRECT: This distractor is credible because a low failure of CV-LT-0113 will close CV-MOV-0113A, VCT OUTL ISOL, and open CV-MOV-0113B, RWST TO CHG PUMP SUCTION ISOL effectively borating the RCS. Entry in to 0POP04-TM-0001, Turbine Load Rejection, is credible because turbine load would start to lower from 100% power with Tavg lowering and the Main Turbine in IMP-IN but this would not be a true load rejection in the sense that Tavg normally goes up with a turbine load rejection.

C: INCORRECT: This distractor is credible because a low failure of CV-LT-0112 will close CV-MOV-0112B, VCT OUTL ISOL, and open CV-MOV-0112C, RWST TO CHG PUMP SUCTION ISOL effectively borating the RCS but CV-LT-0112 controls make-up to the VCT and if it was failed low a continuous auto make-up would occur.

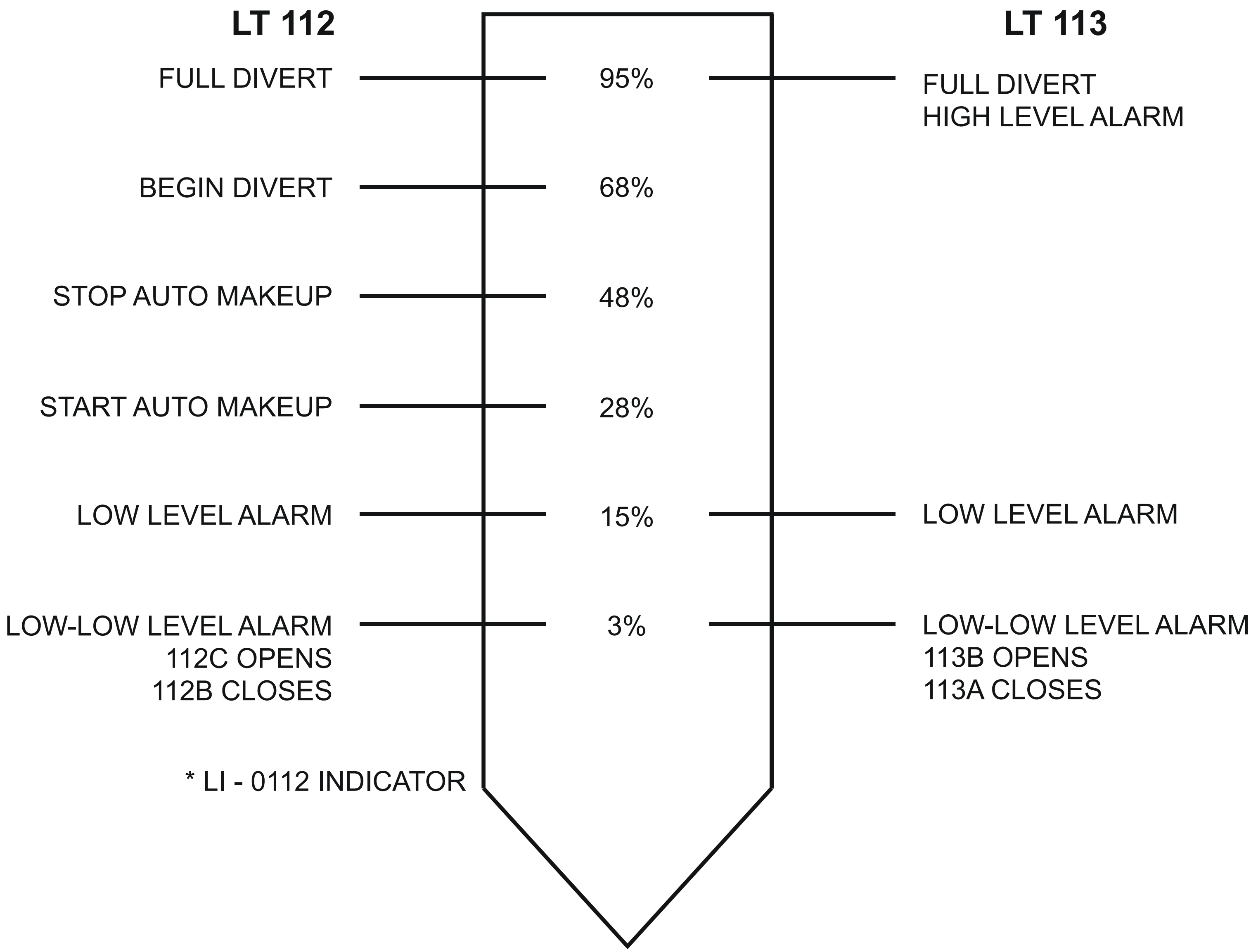
D: INCORRECT: This distractor is credible because a low failure of CV-LT-0112 will close CV-MOV-0112B, VCT OUTL ISOL, and open CV-MOV-0112C, RWST TO CHG PUMP SUCTION ISOL effectively borating the RCS but CV-LT-0112 controls make-up to the VCT and if it was failed low a continuous auto make-up would occur. Entry in to 0POP04-TM-0001, Turbine Load Rejection, is credible because turbine load would start to lower from 100% power with Tavg lowering and the Main Turbine in IMP-IN but this would not be a true load rejection in the sense that Tavg normally goes up with a turbine load rejection.

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to evaluate the given condition and then enter the correct procedure.

VCT LEVEL INSTRUMENTS LT-112 & LT-113



VCT LEVEL LO-LO

Automatic Actions:

1) IF LT-0112 output drops below setpoint (3%), THEN "RWST ISOL MOV-0112C" will OPEN AND "OUTL ISOL MOV-0112B" will CLOSE.
2) IF LT-0113 output drops below setpoint (3%), THEN "RWST ISOL MOV-0113B" will OPEN AND "OUTL ISOL MOV-0113A" will CLOSE.

Immediate Actions: None

Subsequent Actions:

_____ 1) IF actual VCT level is less than 3%, THEN ENSURE Charging pump suction aligns to RWST.

_____ 2) CHECK the following Plant Computer points to identify a failed VCT level transmitter:

_____ a) CVLA0112

_____ b) CVLA0113

_____ 3) IF actual VCT level is less than 3%, THEN PERFORM the following:

_____ a) ENSURE Reactor Makeup system started in AUTO.

_____ b) IF RCS Makeup can NOT be started in AUTO, THEN Manually INITIATE makeup to the RCS per 0POP02-CV-0001, Makeup to the Reactor Coolant System.

_____ 4) IF VCT low level is due to excessive RCS leakage, THEN **GO TO** the leakage procedure appropriate for plant conditions:

- 0POP04-RC-0003, Excessive RCS Leakage
- 0POP04-RC-0006, Shutdown LOCA
- 0POP04-RC-0007, Mode 5 Or Mode 6 LOCA With The Reactor Vessel Head On

04M8-F-2

VCT LEVEL LO-LO

VCT LEVEL LO-LO

Subsequent Actions: (continued) _____ 5) IF a VCT level transmitter (LT-0112 or LT-0113) is failed less than or equal to 3%, THEN PERFORM the following:

NOTE

Holding the VCT outlet isolation valve handswitch in the OPEN position bypasses an automatic SI function to close the valve.

_____ a) OPEN affected VCT outlet isolation valve by taking the VCT outlet isolation valve handswitch to OPEN AND HOLDING it in the open position until the action of either substep b) or c) is complete. {CP004}

- “VCT OUTL ISOL MOV-0112B”
- “VCT OUTL ISOL MOV-0113A”

_____ b) DISPATCH an Operator to OPEN affected VCT outlet isolation MCC breaker:

- “VCT OUTLET ISOL 1(2)-CV-MOV-0112B”; MCC E1C1(E2C1)/G3
- “VCT OUTLET ISOL 1(2)-CV-MOV-0113A”; MCC E1B2(E2B2)/E1

Exam Bank No.: 2164 **Last used on an NRC exam:** Never

SRO Sequence Number: 92

In regards to the Pressurizer Pressure Reactor Trip Setpoint, which of the following describes the Technical Specification Bases for the Setpoint?

A. Prevent water relief through the Pressurizer Safety Valves.

B. Protect against Departure from Nucleate Boiling and Reactor Coolant System over pressure.

C. Prevent Fuel Pellet melting and greater than 1% cladding strain.

D. Protect against consequences of a power excursion from all power levels.

Answer: B Protect against Departure from Nucleate Boiling and Reactor Coolant System over pressure.

Exam Bank No.: 2164

K/A Catalog Number: 010 G2.2.25 **Tier:** 2 **Group/Category:** 1

SRO Importance: 4.2 **10CFR Reference or SRO Objective:** 55.43(b)(2)

Pressurizer Pressure Control: Knowledge of the basis in Technical Specifications for limiting conditions of operations and safety limits.

STP Lesson: LOT 201.14 **Objective Number:** 8559

STATE the function of the pressurizer pressure and level control system components, controls and instrumentation.

Reference: LOT 201.14 Lesson on PZR Pressure and Level Control and TS 2.2.1 Reactor Trip System Instrumentation Setpoints Basis.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible because it is the bases for a different reactor trip (pressurizer level)

B: CORRECT: Preventing DNB is the basis for PZR pressure low Reator Trip setpoint and protecting the RCS from over pressure is the basis for the high Reactor Trip setpoint.

C: INCORRECT: Credible because it is the bases for a different reactor trip (overpower delta-T)

D: INCORRECT: Credible because it is the bases for a different reactor trip (power range high flux)

Question Level: F **Question Difficulty** 3

Justification:

The Unit Supervisor needs to have knowledge of Technical Specification basis.

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations to mitigate the consequences of a reactivity excursion from all power levels.

The Low Setpoint trip may be manually blocked above P-10 (a power level of approximately 10% of RATED THERMAL POWER) and is automatically reinstated below the P-10 Setpoint.

Power Range, Neutron Flux, High Rates

The Power Range High Positive Rate reactor trip provides protection against rapid flux increases which are characteristic of a rupture of a control rod drive housing or uncontrolled RCCA bank withdrawal at power. This trip complements the Power Range Neutron Flux High and Low reactor trip to ensure that the criteria are met for rod ejection from mid power. This trip, in conjunction with the Pressurizer Pressure High reactor trip ensures protection against reactor coolant system overpressurization for the uncontrolled RCCA bank withdrawal at power event.

Intermediate and Source Range, Neutron Flux

The Intermediate and Source Range, Neutron Flux trips provide core protection during reactor startup to mitigate the consequences of an uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the Low Setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a Reactor trip at about 10^5 counts per second unless manually blocked when P-6 becomes active. The Source Range channels are automatically blocked above P-10. The Intermediate Range channels will initiate a Reactor trip at a current level equivalent to approximately 25% of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors, and pressure is within the range between the Pressurizer High and Low Pressure trips. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature-induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in the Core Operating Limits Report. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip is automatically reduced according to the notations in Table 2.2-1 and as specified in the Core Operating Limits Report.

Overpower ΔT

The Overpower ΔT trip provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The Setpoint is automatically varied with: (1) coolant temperature to correct for temperature-induced changes in density and heat capacity of water, and (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, to ensure that the allowable heat generation rate (kW/ft) is not exceeded. The Overpower ΔT trip provides protection to mitigate the consequences of various size steam breaks as reported in WCAP-9226, "Reactor Core Response to Excessive-Secondary Steam Releases."

Pressurizer Pressure

In each of the pressurizer pressure channels, there are two independent bistables, each with its own trip setting to provide for a High and Low Pressure trip thus limiting the pressure range in which reactor operation is permitted. The Low Setpoint trip protects against low pressure which could lead to DNB by tripping the reactor in the event of a loss of reactor coolant pressure.

On decreasing power, the Low Setpoint trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

The High Setpoint trip functions in conjunction with the pressurizer relief and safety valves to protect the Reactor Coolant System against system overpressure.

Pressurizer Water Level

The Pressurizer High Water Level trip is provided to prevent water relief through the pressurizer safety valves. On decreasing power, the Pressurizer High Water Level trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10% of full power equivalent); and on increasing power, automatically reinstated by P-7.

Reactor Coolant Flow

The Low Reactor Coolant Flow trips provide core protection to prevent DNB by mitigating the consequences of a loss of flow resulting from the loss of one or more reactor coolant pumps.

On increasing power above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine impulse chamber pressure at approximately 10% of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below approximately 92% of nominal full loop flow. Above P-8 (a power level of approximately 40% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below approximately 92% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7, an automatic Reactor trip will occur on low reactor coolant flow in more than one loop, and below P-7 the trip function is automatically blocked. The value for loop design flow is the analytical value consistent with the thermal design flow assumed in the DNB analysis.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified Setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Exam Bank No.: 2174 **Last used on an NRC exam:** Never

SRO Sequence Number: 93

Unit 1 is at full power operations.

Annunciator "125V DC SYSTEM E1A11 TRBL" is received and the crew sends a Plant Operator to the Class 1E 125V DC E1A11 Bus.

The Plant Operator reports the following:

- Bus voltage reads 129 VDC
- Negative voltage reads 25 VDC
- Positive voltage reads 105 VDC

Which of the following describes the malfunction and the procedure the Unit Supervisor should use to mitigate the consequences?

	MALFUNCTION	PROCEDURE
A.	Class 1E 125 VDC Bus has a ground.	Use 0POP02-EE-0001, ESF (Class1E) DC Distribution System, to de-energize the Bus and apply appropriate Tech Specs.
B.	Class 1E 125 VDC Bus has a ground.	Use 0POP01-ZO-0009, Ground Isolation, to attempt to isolate the ground fault.
C.	Class 1E 125 VDC Bus has high current.	Use 0POP02-EE-0001, ESF (Class1E) DC Distribution System, to place a second battery charger in service.
D.	Class 1E 125 VDC Bus has high current.	Use 0POP01-ZO-0009, Ground Isolation, to de-energize unnecessary loads.

Answer: B Class 1E125 VDC Bus has a ground. Use 0POP01-ZO-0009, Ground Isolation, to attempt to isolate the ground fault.

Exam Bank No.: 2174

K/A Catalog Number: 063 A2.01 **Tier:** 2 **Group/Category:** 1

SRO Importance: 3.2 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to (a) predict the impacts of the following malfunctions or operations on the DC electrical systems; and (b) based on those predictions, control or mitgate the consequences of those malfunctions or operations: Grounds

STP Lesson: LOT 201.37 **Objective Number:** 92986

DESCRIBE the local and MCR instrumentation available to monitor the Class 1E 125 VDC System.

Reference: LOT 201.37 and Annunciator Response, 1POP09-AN-03M2, Window A-1, 125V DC SYSTEM E1A11 TRBL.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: De-energizing the bus is credible because it will alleviate the ground, however this action is not proceduralized or desired in this condition.

B: CORRECT: A difference in negative and positive bus voltage is an indication of a ground. Normal positive bus volts is 65 and negative bus volts is 65. The Unit Supervisor should try to locate the source of the ground using the indicated procedure.

C: INCORRECT: Credible because high current may lower voltage, but will not cause a difference in positive and negative readings. Additional charger capacity could prevent damage to batteries in an overcurrent condition, however these chargers are not designed to operate in parallel.

D: INCORRECT: Credible because high current may lower voltage, but will not cause a difference in positive and negative readings. ZO-09 provides guidance on de-energizing loads, however it would be an incorrect application of the procedure.

Question Level: H **Question Difficulty** 2

Justification:

The Unit Supervisor has to evaluate the given condition and select the correct malfunction and Procedure.

125V DC SYSTEM E1A11 TRBL

Subsequent Actions: (Continued)

_____ 11) IF bus current is greater than or equal to 200 amps OR battery charger current is greater than or equal to 300 amps, THEN CONTACT Electrical Maintenance for assistance in determining the cause.

_____ 12) IF a ground is indicated, THEN DIRECT an Operator to isolate equipment as directed by the Unit/Shift Supervisor to attempt to isolate the fault. REFER TO 0POP01-ZO-0009, Ground Isolation.

_____ 13) TAKE appropriate actions per Technical Specifications 3.8.1.1, 3.8.1.2, 3.8.1.3, 3.8.2.1, 3.8.2.2, 3.8.3.1, 3.8.3.2, TRM 3.8.1, and 3.8.2.

_____ 14) INITIATE a CR to document faulted conditions and to repair the faulted condition.

_____ 15) WHEN desired to return system to normal after repair, THEN PERFORM applicable section of 0POP02-EE-0001, ESF (Class1E) DC Distribution System **OR** 0POP02-AE-0004, 120 VAC ESF Vital Distribution Power Supplies.

Probable Causes:

1) Loss of battery charger AC input power
2) Battery charger failure
3) Any E1A11 switchboard breaker tripped
4) E1A11 switchboard trouble
5) Instrument failure
6) System ground

Exam Bank No.: 2177 **Last used on an NRC exam:** Never

SRO Sequence Number: 94

Which of the following requires Core Load Supervisor approval prior to performing?

During an outage and while offloading the Reactor Core…

A. the FHB Upender Operator uses the “PROX SWITCH BYPASS” key switch to operate the FHB upender because the “UPENDER CLEAR” light will not illuminate.

B. the FHB Upender Operator positions the “TRAVERSE RX/POOL” selector switch to “RX” position to move the carriage from the FHB to Containment.

C. the Fuel Handling Machine Operator lowers a fuel assembly into a Region 2 Fuel Cell.

D. the Fuel Handling Machine Operator unlatches a fuel assembly after being placed in a SFP Region 1 Fuel Cell.

Answer: A the FHB Upender Operator uses the"PROX SWITCH BYPASS" key switch to operate the FHB upender because the "UPENDER CLEAR" light will not illuminate.

Exam Bank No.: 2177

K/A Catalog Number: 034 G2.1.42 **Tier:** 2 **Group/Category:** 2

SRO Importance: 3.4 **10CFR Reference or SRO Objective:** 55.43(b)(7)

Fuel Handling Equipment: Knowledge of new and spent fuel movement procedures.

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: LOT 201.43 Introduction to Refueling and 0POP08-FH-0002, Fuel Handling Machine, and 0POP08-FH-0003, Fuel Transfer System.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: Per procedural guidance the Core Load Supervisor, during refueling operations, has to approve any bypass operation of Fuel Handling equipment.

B: INCORRECT: This is a credible distractor because moving the Fuel Transfer system carriage requires two verifications prior to moving; the Upender Frame Down light must be illuminated and the operator must visually verify that the frame is down but this action does not require Core Load Supervisor permission.

C: INCORRECT: This is a credible distractor because independent verification is required to lower a fuel assemble, but not from the Core Load Supervisor.

D: INCORRECT: This is a credible distractor because Step Verifier agreement is required to unlatch a fuel assemble, but not from the Core Load Supervisor.

Question Level: F **Question Difficulty** 3

Justification:

The Unit Supervisor needs to have knowledge of Fuel Handling Procedure precautions and notes.

4.0 Notes and Precautions

4.1 Precautions specified in applicable Radiation Work Permit SHALL be observed.

4.2 The Fuel Transfer Canal and Refueling Cavity SHALL be governed by Housekeeping Zone 3 guidelines per 0PGP03-ZA-0098, Station Housekeeping.

4.3 The areas above the spent fuel pool, fuel transfer canals, an uncovered new fuel vault, the reactor vessel, and the in-containment storage area (with stored fuel or during movement of fuel in the RCB), SHALL be considered a FMEA Level 1 in accordance with 0PGP03-ZA-0014, Foreign Material Exclusion Program.

4.4 0PEP02-ZM-0009, Spent Fuel Pool Storage and Work, provides requirements and guidelines for all work conducted in the SFP, ICSA, the Reactor Cavity, and the fuel transfer canals when filled with water. These requirements and guidelines apply to all individuals who perform work in the above areas.

4.5 IF moving fuel or other irradiated components, THEN fuel transfer equipment SHALL **NOT** be operated until Transfer Tube Quick Opening Hatch AND Fuel Transfer Tube Gate Valve are opened.

4.6 Key operated interlock bypass switches are administratively controlled.

4.7 IF moving fuel or other irradiated components, THEN key operated interlock bypass switches SHALL **ONLY** be used with Core Loading Supervisor's permission.

4.8 Control console heaters are wired independently of main power input. IF control consoles are required to be opened for maintenance activities with the heater circuit energized, THEN personnel must take precautions to avoid electrical shock hazards. All work inside the control consoles will be performed in accordance with an approved work package.

4.9 WHEN fuel transfer activities are in progress, THEN communications SHALL be maintained between the Refueling Machine, Fuel Handling Machine, RCB Control Console, FHB Control Console, and Main Control Room. (TRM 3.9.5 and Reference 2.10)

4.10 Prior to lifting a fuel assembly from either Upender, ensure Upender is in vertical position.

4.11 WHEN irradiated fuel is being transported through Fuel Transfer Tube, THEN high radiation conditions may exist in FHB penetration space in vicinity of Fuel Transfer Tube. Access to this area SHALL be controlled per Form 3 of 0PRP07-ZR-0009, Performance of High Exposure Work. (Reference 2.8)

4.12 This procedure applies to transfer of irradiated material from RCB to FHB, provided material has been configured or packaged so it can be placed in the Fuel Transfer System Fuel Container. In such cases the words "fuel assembly" in this procedure can be construed to mean, "irradiated material".

Exam Bank No.: 2184 **Last used on an NRC exam:** Never

SRO Sequence Number: 95

Unit 2 is at 100% Power.

The Reactor Trip Switchgear loses DC control power from Class 1E 125VDC E1A11.

Subsequently SSPS receives an automatic Reactor Trip signal.

Which of the following (1) describes how the Reactor Trip Breakers are affected and (2) the direction the Unit Supervisor should give to mitigate the consequences of the malfunction?

A. (1) Only Reactor Trip Breaker ‘R’ Shunt Trip Coil loses power.
(2) Enter 0POP05-EO-FRS1, Response to Nuclear Power Generation – ATWS.

B. (1) Reactor Trip Breaker ‘R’ and ‘S’ Shunt Trip Coil loses power.
(2) Enter 0POP05-EO-FRS1, Response to Nuclear Power Generation - ATWS.

C. (1) Reactor Trip Breaker ‘R’ and ‘S’ Shunt Trip Coil loses power.
(2) Enter 0POP05-EO-EO00, Reactor Trip or Safety Injection.

D. (1) Only Reactor Trip Breaker ‘R’ Shunt Trip Coil loses power.
(2) Enter 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Answer: D (1) Only Reactor Trip Breaker ‘R’ Shunt Trip Coil loses power.
(2) Enter 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Exam Bank No.: 2184

K/A Catalog Number: 012 A2.07 **Tier:** 2 **Group/Category:** 1

SRO Importance: 3.7 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to (a) predict the impacts of the following malfunctions or operations on the RPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequeces of those malfunctions or operations: Loss of dc control power

STP Lesson: LOT 201.20 **Objective Number:** 507227

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

Reference: LOT 201.20 Lesson on SSPS and LOT 201.37 Lesson on Class 1E 125VDC

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Part 2 is credible because all other remotely operated breakers will not work if control power is lost, however trip breakers have two independent means of opening and will still open when DC power is lost.

B: INCORRECT: Part 1 is credible because both reactor trip breakers receive control power from Class 1E DC, but not from the same source. Part 2 is credible because all other remotely operated breakers will not work if control power is lost, however trip breakers have two independent means of opening and will still open when DC power is lost.

C: INCORRECT: Part 1 is credible because both reactor trip breakers receive control power from Class 1E DC, but not from the same source.

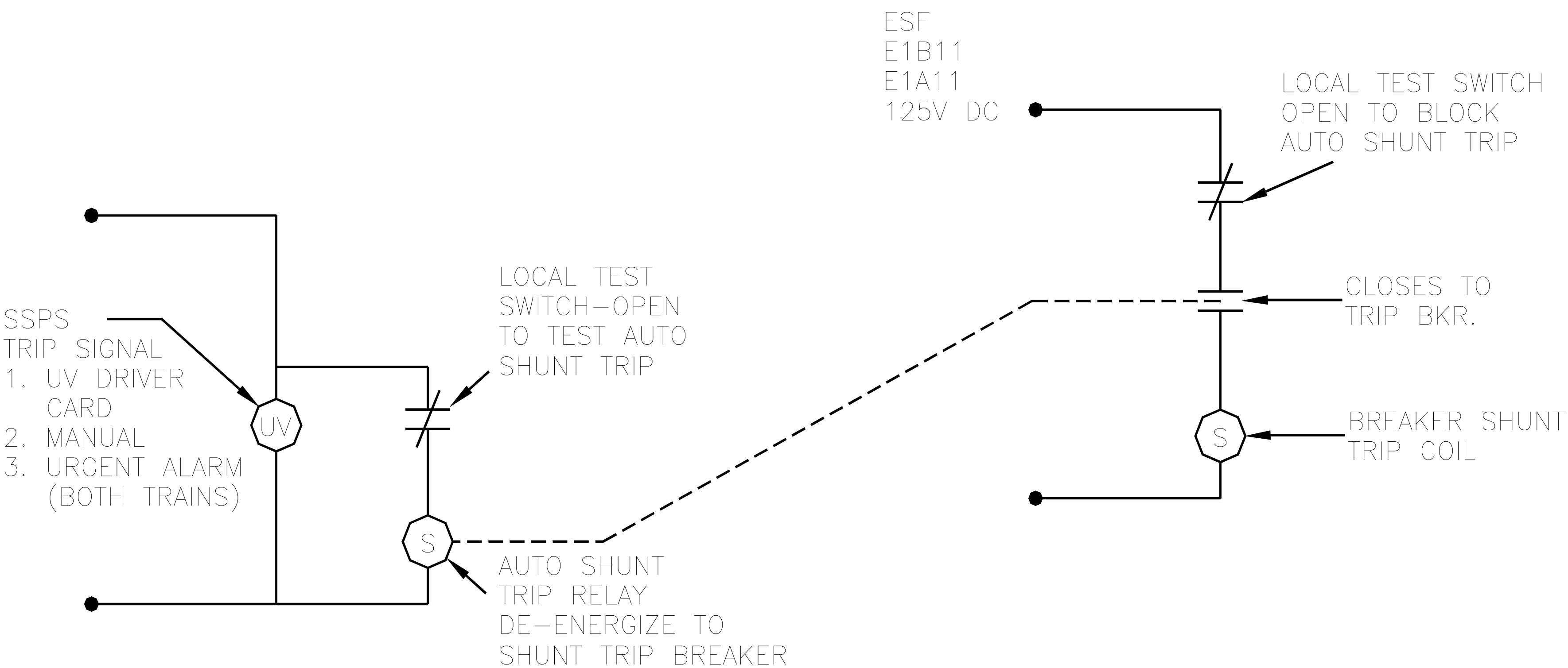
D: CORRECT: Reactor Trip Breaker 'R' Shunt Trip Coil gets power from Class 1E 125 VDC E1A11. Entering 0POP05-EO-EO00, Reactor Trip or SI is correct because the 48 volt dc uv coil would still trip both Reator Trip Breakers open.

Question Level: H **Question Difficulty** 4

Justification:

The Unit Supervisor must evaluate the given condition and determine the correct procedure to implement. Must also have basic knowledge of how the RTBs operate.

REACTOR TRIP BREAKER UV AND SHUNT TRIP ARRANGEMENT



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A-Train Loads

- Reactor Trip Breaker Control Power
- ESF DG “A” Field Flash
- PL-139A/39A (DG Control Panels)
- 4160/480 VAC (ESF) breaker control power
- Load Sequencer A
- Inverters 001 and 1201
- Pzr PORV 655A

B-Train Loads

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

Exam Bank No.: 2185 **Last used on an NRC exam:** Never

SRO Sequence Number: 96

Unit 1 is in a Site Area Emergency based on the following:

- A Core Cooling Orange Path has been in effect for 20 minutes.
- Core Exit Thermocouples are 715ºF and slowly rising.
- RCS Plenum level is 20%.

Which of the following parameters would cause an escalation to a General Emergency?

NOTE: Consider each of the following separately.

A. Reactor Coolant System Activity (DEI) is reported at 350µCi/gm.

B. Reactor Coolant Failed Fuel Monitor, RT-8039, is reading 900µCi/ml.

C. Containment Pressure is 25psig.

D. Containment Hatch Monitor is reading 200mR/hr.

Answer: A Reactor Coolant System Activity (DEI) is reported at 350µCi/gm.

Exam Bank No.: 2185

K/A Catalog Number: 002 G2.4.41 **Tier:** 2 **Group/Category:** 2

SRO Importance: 4.6 **10CFR Reference or SRO Objective:** Objective SRO 74026

Reactor Coolant: Knowledge of the emergency action level thresholds and classifications.

STP Lesson: LOT 803.14 **Objective Number:** 74026

Given an emergency condition and a copy of the emergency classification tables from 0ERP01-ZV-IN01, Emergency Classification, CLASSIFY the emergency condition.

Reference: LOT 803.14 - 0ERP01-ZV-IN01, Emergency Classification

Attached Reference ☑ **Attachment:** 0ERP01-ZV-IN01, Emergency Classification, Addendum 1, Emergency Classification Tables, Page 2 and 3.

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: With the given conditions the Fission Product Barrier Degradation totals 8 points. 3 points for potential loss of fuel clad due to Core Cooling Orange Path or RCS Plenum level at 20% or CETs at 715 degrees F. 4 points for loss of RCS due to Core Cooling Yellow with subcooling less than 0 degrees F. (Core Cooling Orange and CETs at 715 degrees F would satisfy this) 1 point for potential loss of Containment due to Core Cooling Orange greater than 15 minutes. With RCS DEI at 350 uCi/gm fuel clad would go from potential loss to loss. (3 points to 4 points) This would make the Fission Product Barrier total 9 points and thus raise the E-Plan declaration to a General Emergency.

B: INCORRECT: This is a credible distractor because although the failed fuel monitor reading of 900uCi/ml is greater than the limit of 870uCi/ml this represents on a potential loss of the fuel clad which has already been identified.

C: INCORRECT: This is a credible distractor because athough Containment pressure of 25 psig is considerably high, Containment pressure alone could not cause an escalation to a GE because it would only represent a potential loss of Containment which has already been identified.

D: INCORRECT: This is a credible distractor because although an elevated reading on the Containment Hatch Rad Monitor could cause an escalation in the E-Plan classification the reading would have to go above 222mR/hr.

Question Level: H **Question Difficulty** 3

Justification:

The SRO has to evaluate the condition given and determine which parameter would cause an escalation of emergency classification level to a GE.

RECOGNITION CATEGORY F
FISSION PRODUCT BARRIER DEGRADATION
INITIATING CONDITION MATRIX

Determine which combination of the three barriers are lost or have a potential loss and use the following matrix to classify the event. Also, an event (or multiple events) could occur which result in the conclusion that the loss or potential loss is IMMINENT (within 1 to 2 hours). In this IMMINENT loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT (1-2)	ALERT (3-4)	SITE AREA EMERGENCY (5-8)	GENERAL EMERGENCY (9-10)
FU1 ANY Loss or ANY Potential Loss of Containment **FU2** Fuel Clad Degradation See SU6 **FU3** RCS Leakage - See SU7	**FA1** ANY Loss or ANY Potential Loss of Fuel Clad or RCS	**FS1** Loss of BOTH Fuel Clad and RCS OR Potential Loss of BOTH Fuel Clad and RCS **OR** Potential Loss of EITHER Fuel Clad or RCS **AND** Loss of ANY Additional Barrier	**FG1** Loss of ANY Two Barriers AND Potential Loss or Loss of Third Barrier

Operating Modes 1 through 4

Note: 1. At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from General Emergency.

2. The ability to escalate to higher emergency classes as an event degrades must be maintained. RCS leakage steadily increasing would represent an increasing risk to public health and safety.

<u>Determination of Emergency Classification Level</u>
Select values from the top of the columns on the next page, which describe specific Fission Product Barrier degradation. Select the higher value that applies from each barrier. Add the values to arrive at the total challenge to the Fission Product Barriers. The emergency classification is determined from the range of values shown in parentheses in the table above.

RECOGNITION CATEGORY F
FISSION PRODUCT BARRIER DEGRADATION
INITIATING CONDITION MATRIX

	FUEL CLAD		RCS		CONTAINMENT	
EAL	POTENTIAL LOSS (3)	LOSS (4)	POTENTIAL LOSS (3)	LOSS (4)	POTENTIAL LOSS (1)	LOSS (2)
1	CSF Core Cooling - Orange **OR** Heat Sink - Red[2]	CSF Core Cooling - Red	CSF RCS Integrity – Red **OR** Heat Sink - Red[2]	CSF Core Cooling - Yellow with subcooling < 0 °F	CSF Containment - Red **OR** Core Cooling - Orange > 15 min.	–
2	RCS Activity Failed Fuel Monitor, RT-8039, equal to or greater than 870 μCi/ml	RCS Activity Dose Equivalent Iodine greater than 300 μCi/gm	RCS Leak Rate Unisolable leak exceeding the capacity of one centrifugal charging pump in the normal charging mode.	RCS Leak Rate Leak rate greater than CVCS System's ability to maintain RCS inventory as indicated by loss of RCS subcooling.	Containment Pressure Greater than 6% hydrogen concentration in containment **OR** Containment pressure greater than 9.5 psig with neither containment spray nor RCFC running.	Containment Pressure Initial increase followed by rapid unexplained decrease **OR** Containment pressure or sump level not increasing as expected with LOCA conditions.
3	Core Exit Thermocouple ≥ 708°F	Core Exit Thermocouple 1200°F	SG Tube Rupture SG Tube has ruptured and the primary to secondary leak rate is greater than the capacity of one centrifugal charging pump.	SG Tube Rupture SG Tube is ruptured and has a non-isolable secondary steam release	–	SG Tube Leak Primary to secondary leakage greater than 150 gpd through any one steam generator with direct secondary side leakage to atmosphere
4	Reactor Vessel Water Level Plenum level less then 20%	–	–	–	Containment Bypass VALID increase in reading on area or ventilation monitors in areas adjacent to the containment boundary with a known LOCA inside containment.	Containment Isolation Containment isolation signal **AND** Valves not closed **AND** A pathway to the environment exists.
5	–	RCB Rad Monitor RT-8050 or RT-8051 greater than 100 R/hr **OR** Hatch Monitor greater than 222 mR/hr	–	RCB Rad Monitor RT-8050 or RT-8051 greater than 100 R/hr **OR** Hatch Monitor greater than 222 mR/hr	RCB Rad Monitor RT-8050 or RT-8051 greater than 1,000 R/hr **OR** Hatch Monitor greater than 2,222 mR/hr	–

Note: 1. The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Containment Barrier. Unusual Event Initiating Conditions (ICs) associated with RCS and Fuel Clad barriers are addressed under SU6 and SU7.

2. CSF indicators must be valid; outside the immediate control of the operator.

Exam Bank No.: 2187 **Last used on an NRC exam:** Never

SRO Sequence Number: 97

A Site Area Emergency has been declared in Unit 1. The TSC, EOF and JIC have not been activated yet.

An Inside Containment Isolation Valve needs to be manually closed to stop a radiological release that is affecting the owner controlled area. It is estimated that it will take a worker 15 minutes to close the valve once in the area. The dose rate in the area of the valve is estimated at 25 REM/Hour.

In accordance with 0ERP01-ZV-IN06, Radiological Exposure Guidelines, who at the minimum should provide approval to perform this task?

A. Emergency Director Only

B. Emergency Director and the Worker

C. The Worker Only

D. Acting Radiological Manager and the Worker

Answer: B Emergency Director and the Worker

Exam Bank No.: 2187

K/A Catalog Number: 103 G2.4.38 **Tier:** 2 **Group/Category:** 1

SRO Importance: 4.4 **10CFR Reference or SRO Objective:** 55.43(b)(4)

Containment: Ability to take actions called for in the facility emergency plan, including supporting or acting as emergency coordinator if required.

STP Lesson: LOT 803.14 **Objective Number:** 65180

Given a description of responsibilities related to an Emergency Response Organization position that interfaces with the Emergency Director, DETERMINE the responsible individual by title.

Reference: LOT 803.14 - 0ERP01-ZV-SH01, Shift Manager - Responsibilities which include directing control room response to mitigate the emergency condition.

Attached Reference ☑ **Attachment:** 0ERP01-ZV-IN06, Radiological Exposure Guidelines - 7 Page Procedure

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Credible beacause the Emergency Director must sign, however since 10CFR20 limits are being exceeded, the worker must sign too.

B: CORRECT: Since the exposure to the worker will exceed 5 REM then the Worker and the Emergency Director are required to sign for approval. See Data Sheet 1 in the Procedure.

C: INCORRECT: Credible because the exposure to the worker will exceed 5 REM and the Worker is required to sign but the Emergency Director is always required to sign.

D: INCORRECT: Credible because the Acting Radiological Manager has procedural responsibilities for requesting and tracking approvals, however the procedure does not give them signature authority for the ED.

Question Level: H **Question Difficulty** 3

Justification:

The SRO has to evaluate the condition and then use the procedure to determine authorization for the exposure approval.

Name: ________________________________ SSN: ____

Date/Time: ____________________________ Discipline Group: ____________________________

Reason for Dose Extension Request: ____

____

Current exposure to date: ___________ Rem (TEDE) TLD No.: _____________________________

Exposure Authorized to: ____________________________ Rem (TEDE)

_____________________________________ _____________________________________

*Emergency Worker Emergency Director

* Required to exceed 10CFR20 TEDE exposure limit (5 rem).

Exam Bank No.: 2188 **Last used on an NRC exam:** Never

SRO Sequence Number: 98

Given the following:

- A LOCA has occurred in Unit 2
- Operators are performing 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant
- Only one HHSI pump is available and running
- CETs are 710 °F and rising
- Core voiding is just beginning to occur

Which of the following describes the initial response of the Source Range detectors to the core voiding and the actions the Unit Supervisor should take to control the voiding?

	NI Response	Actions
A.	Indication will RISE	Enter 0POP05-EO-FRC1, Response to Inadequate Core Cooling and depressurize the RCS using a Pressurizer PORV to 255 psig to allow accumulators to inject.
B.	Indication will RISE	Enter 0POP05-EO-FRC2, Response to Degraded Core Cooling and start all available charging pumps to raise RCS inventory.
C.	Indication will LOWER	Enter 0POP05-EO-FRC1, Response to Inadequate Core Cooling and depressurize the RCS using a Pressurizer PORV to 255 psig to allow accumulators to inject.
D.	Indication will LOWER	Enter 0POP05-EO-FRC2, Response to Degraded Core Cooling and start all available charging pumps to raise RCS inventory.

Answer: B Indication will RISE; Enter 0POP05-EO-FRC2, Response to Degraded Core Cooling and start all available charging pumps to raise RCS inventory.

Exam Bank No.: 2188

K/A Catalog Number: 015 A2.05 **Tier:** 2 **Group/Category:** 2

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to predict the impacts of the following malfunctions or operations on the NIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Core void formation.

STP Lesson: LOT 502.09 **Objective Number:** 50384

DESCRIBE the effects on reactor kinetics of coolant voiding in the core region, and relate the Excore Nuclear Instrumentation System response to voiding.

Reference: LOT502.09 handout page 39, 0POP05-EO-FRC2 step 2 RNO

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Procedure is credible because FRC1 will be performed if temperature continues to rise. Action is credible because FRC1 does inject accumulators, but by depressurizing SGs.

B: CORRECT: The beginning of core voiding during a LOCA will cause excore NIs to start reading higher due to more neutrons leaking out of the core. With CETs above 708 and not all SI pumps running, FRC2 will be entered and all available charging pumps started to help raise RCS inventory.

C: INCORRECT: NI response is credible because voiding reduces Keff by reducing moderation which would cause NIs to lower (this effect is overcome by increased leakage though). Procedure is credible because FRC1 will be performed if temperature continues to rise. Action is credible because FRC1 does inject accumulators, but by depressurizing SGs.

D: INCORRECT: NI response is credible because voiding reduces Keff by reducing moderation which would cause NIs to lower (this effect is overcome by increased leakage though).

Question Level: H **Question Difficulty** 3

Justification:

The Unit Supervisor has to have knowledge of the operation and effects on nuclear instrumentation and has to evaluate the given condition to determine the procedure to use to mitigate the accident.

is less attenuation of neutron flux in the downcomer, and the transmission ratio increases.

In fact, as the homogeneous void fraction in the core region increases from 0 to 100 percent, the transmission ratio increases by several orders of magnitude (from four to six orders of magnitude, depending upon the core model and method of calculation used). The increase in transmission ratio causes the excore detector response to increase with homogeneous voiding.

Similarities and Variations in Analysis Results

The results of all excore detector response analyses performed following the TMI-2 accident were consistent in the following respect: For homogeneous void fractions of up to about 80 percent, detector response increases (that is, the count rate goes up) continuously.

For homogeneous void fractions in excess of about 80 percent, the detector response results reported in various analyses are not entirely consistent. Some analyses show detector response leveling off at a void fraction of about 80 percent (fig. 3-1.11). Others show detector response continuously increasing for higher void fractions, all the way up to 100 percent voids (fig. 3-1.12).

In all of the analyses, increases in detector response are attributed to increases in the transmission ratio, which in turn are attributed to decreases in downcomer fluid density. The differences in results for homogeneous void fractions greater than 80 percent voids can be attributed to the following differences in the analyses:

- Calculated effects of voiding on k_{eff}
- Calculated effects of voiding on core source strength

All analyses show k_{eff} decreasing as voiding increases. However, some show k_{eff} decreasing more than others, especially at very high void fractions. Similarly, all analyses show core source strength decreasing as voiding increases. However, some show source strength decreasing more than others.

The analyses showing detector response leveling off at very high void fractions are based on comparatively low values for k_{eff} and source strength at high void fractions. In these analyses, the reduction in core neutron population that occurs at very high void fractions counterbalances the increased transmission ratio. Although the neutron leakage probability increases with voiding, there are fewer neutrons available to leak. So the detector response levels off.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- **Core Exit T/Cs GREATER THAN 708°F represents a potential SAE per 0ERP01-ZV-IN01, EMERGENCY CLASSIFICATION. (Matrix)**
- **Normal conditions for running RCPs are desired, but RCPs should NOT be tripped if normal conditions can NOT be established or maintained.**
- **Foldout CIP page should be open.**

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___1	**VERIFY SI Valve Alignment - PROPER EMERGENCY ALIGNMENT**	Manually ALIGN valves.
___2	**VERIFY SI Flow In ALL 3 Trains:**	
	___a. HHSI pump flow - INDICATED In ALL 3 Trains	a. TRY TO establish SI flow: 1) START HHSI pumps. 2) ALIGN valves to establish HHSI flow. 3) <u>IF</u> HHSI flow has been established in ALL 3 trains, <u>THEN</u> GO TO Step 2 b. **RNO Continued on Next Page**

Step 2 continued on next page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 2 continued from previous page.

4) TRY TO establish maximum charging flow:

   a) RESET SI.

   b) RESET Containment Isolation Phase A.

   c) ALIGN charging pump suction to RWST:

      1) OPEN charging pump suction valves from RWST.

      2) CLOSE VCT outlet valves.

   d) IF at least one RWST suction valve can NOT be opened, THEN DISPATCH operator to open charging pump suction valve.

      (10 ft MAB Charging Pump Valve RM 044)

      "1(2)-CV-MOV-0112C"
      "CVCS CHARGING PUMPS"
      "SUCTION FROM RWST"
      "MOV OPERATOR"

   e) IF at least one VCT outlet valve can NOT be closed, THEN DISPATCH operator to close at least one of the VCT outlet isolation valves

      (41 ft MAB CVCS valve RM 226)

      o "1(2)-CV-MOV-0112B"
        "CVCS VOLUME CONTROL TANK"
        "OUTLET MOV OPERATOR"

      o "1(2)-CV-MOV-0113A"
        "CVCS VOLUME CONTROL TANK"
        "OUTLET MOV OPERATOR"

Step 2 continued on next page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 2 continued from previous page.

f) ENSURE charging flow control valve is closed.

g) IF charging flow control valve will NOT close, THEN ESTABLISH maximum charging flow per ADDENDUM 1, ESTABLISHING ALTERNATE MAXIMUM CHARGING FLOW CONTROL.

h) OPEN the recirculation valve for the CCP(s) to be started.

i) START all available CCPs.

j) ENSURE CCPs discharge valves open.

k) ENSURE normal or alternate charging isolation valve open.

l) OPEN charging OCIV.

m) IF charging OCIV will NOT open, THEN DISPATCH operator to open charging OCIV:

(29 ft MAB RM 108C)

"1(2)-CV-MOV-0025"
"CVCS CHARGING ORC"
"CONTAINMENT ISOLATION"
"MOV OPERATOR"

n) OPEN charging flow control valve to obtain maximum flow.

Step 2 continued on next page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 2 continued from previous page.

		o) IF charging flow control valve will NOT open, THEN ESTABLISH maximum charging flow per ADDENDUM 1, ESTABLISHING ALTERNATE MAXIMUM CHARGING FLOW CONTROL. p) CLOSE running CCP(s) recirculation valve(s). q) IF at least one CCP can NOT be started, THEN PERFORM the following: 1) START the PDP. 2) CLOSE "PDP RECIRC HCV-0285".
___	b. RCS pressure - LESS THAN 415 PSIG	b. GO TO Step 3.
___	c. LHSI pump flow - INDICATED In ALL 3 Trains	c. TRY TO establish SI flow: 1) START LHSI pumps. 2) ALIGN valves to establish LHSI flow.

Exam Bank No.: 1604 **Last used on an NRC exam:** Never

SRO Sequence Number: 99

Given the following:

- The crew is performing 0POP05-EO-FRH1, Response to Loss of Secondary Heat Sink, due to an earlier red path on the Heat Sink Critical Safety Function.

Subsequently:

- The STA reports the following current conditions:
  - Subcriticality CSF ORANGE
  - Core Cooling CSF ORANGE
  - Heat Sink CSF YELLOW
  - Integrity CSF RED
  - Containment CSF RED
  - Inventory CSF YELLOW

Which one of the following correctly describes the actions required?

Complete 0POP05-EO-FRH1 and transition to…

A. 0POP05-EO-FRZ1, Response to High Containment Pressure.

B. 0POP05-EO-FRC2, Response to Degraded Core Cooling.

C. 0POP05-EO-FRP1, Response to Imminent Pressurized Thermal Shock Condition.

D. 0POP05-EO-FRS1, Response to Nuclear Power Generation - ATWS

Answer: C 0POP05-EO-FRP1, Response to Imminent Pressurized Thermal Shock Condition.

Exam Bank No.: 1604

K/A Catalog Number: G2.4.14 **Tier:** 3 **Group/Category:** 4

SRO Importance: 4.5 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Knowledge of general guidelines for EOP usage.

STP Lesson: LOT 504.04 **Objective Number:** 92283

Given a set of conditions and the occurrence of a Red, Orange, or Yellow path CSF, STATE the action required per 0POP01-ZA-0018, EOP Users Guide.

Reference: 0POP01-ZA-0018, EOP User's Guide

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified From**

Distractor Justification

A: INCORRECT: Credible because a red condition is a higher priority than the current yellow condition on Heat Sink, but not correct because it is lower in priority than Integrity.

B: INCORRECT: Credible because Core Cooling is a higher priority CSF than the current Heat Sink, but not correct because red overrides orange and Subcriticality is higher priority than Core cooling.

C: CORRECT: Per ZA-0018, a red or orange priority FRP must be completed unless preempted by a higher priority condition. Once complete, transition is made to the next highest priority (color and CSF) condition.

D: INCORRECT: Credible because Subcriticality is a higher priority CSF than the current Heat Sink, but not correct because red overrides orange.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must have knowledge of the hierarchy that exists for critical safety functions and their associated colors and be able to evaluate the given conditions and determine the appropriate action IAW 0POP01-ZA-0018

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.

Exam Bank No.: 1732 **Last used on an NRC exam:** Never

SRO Sequence Number: 100

Unit 1 is operating at 100% power when Chemistry notifies the Control Room of the following results:

- RCS gross activity of 219 microcuries per gram and a newly calculated Ē (E-bar) of 0.2734.

Based on this information, as a minimum, the Unit Supervisor should....

A. Enter 0PGP03-ZO-0012, Plant Systems Chemistry Control, and shutdown as quickly as safe plant operation allows to comply with Tech Spec 3.4.8, RCS Specific Activity.

B. Enter 0PGP03-ZO-0012, Plant Systems Chemistry Control, and immediately take action to reduce power to 50% or less to comply with Tech Spec 3.4.8, RCS Specific Activity.

C. Enter POP04-RC-0001, High Reactor Coolant System Activity, and raise letdown flow in order to pass more coolant through the Demineralizers.

D. Enter POP04-RC-0001, High Reactor Coolant System Activity, and lower letdown flow in order to allow the coolant to remain in the Demineralizers longer.

Answer: C Enter POP04-RC-0001, High Reactor Coolant System Activity, to raise letdown flow in order to pass more coolant through the Demineralizers.

Exam Bank No.: 1732

K/A Catalog Number: G2.1.34 **Tier:** 3 **Group/Category:** 1

SRO Importance: 3.5 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Knowledge of primary and secondary plant chemistry limits.

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

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

Reference: TS 3.4.8; POP04-RC-0001, High Reactor Coolant System Activity

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified From**

Distractor Justification

A: INCORRECT - Credible because RCS activity is listed in the chemistry specification procedure (but does not have an action level so ZO-0012 would not be entered). The action specified follows the required action for a level 3 violation in ZO-0012. The Tech Spec inference is credible because there is a Tech Spec limit of 100/E-bar for RCS activity (which is not exceeded by the given conditions).

B: INCORRECT - Credible because RCS activity is listed in the chemistry specification procedure (but does not have an action level so ZO-0012 would not be entered). The action specified follows the required action for a level 2 violation in ZO-0012. The Tech Spec inference is credible because there is a Tech Spec limit of 100/E-bar for RCS activity (which is not exceeded by the given conditions).

C: CORRECT - Reported activity does not exceed the TS limit (calculated limit is 365.76 microcuries) but is an elevated reading so POP04 entry would be required. The POP04 has the crew raise letdown flow to improve RCS cleanup.

D: INCORRECT - Credible beacause lower flow would mean more contact time with the resin in the demineralizer, however raising flow will actually clean up a given volume faster provided channeling does not occur which is the reason for the maximum flow in the POP04.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must calculate the Tech Spec limit based on the given E-bar to determine if Tech Specs have been exceeded. Then, based on this determination choose the correct procedure response.

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)

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 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	GO TO Step 8.0.

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the reactor coolant shall be limited to:

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/Ē microCuries per gram of gross radioactivity.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval, or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours; and

b. With-the gross specific activity of the reactor coolant greater than 100/Ē microCuries per gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

c. Specification 3.0.4.c is applicable.

MODES 1, 2, 3, 4, and 5:

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/Ē micro-Curies per gram, perform the sampling and analysis requirements of Item 4.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

*With T_{avg} greater than or equal to 500°F.

LOT 19 NRC Exam SRO Reference Package

RECOGNITION CATEGORY F
FISSION PRODUCT BARRIER DEGRADATION
INITIATING CONDITION MATRIX

Determine which combination of the three barriers are lost or have a potential loss and use the following matrix to classify the event. Also, an event (or multiple events) could occur which result in the conclusion that the loss or potential loss is IMMINENT (within 1 to 2 hours). In this IMMINENT loss situation use judgment and classify as if the thresholds are exceeded.

UNUSUAL EVENT (1-2)	ALERT (3-4)	SITE AREA EMERGENCY (5-8)	GENERAL EMERGENCY (9-10)
FU1 ANY Loss or ANY Potential Loss of Containment **FU2** Fuel Clad Degradation See SU6 **FU3** RCS Leakage - See SU7	**FA1** ANY Loss or ANY Potential Loss of Fuel Clad or RCS	**FS1** Loss of BOTH Fuel Clad and RCS OR Potential Loss of BOTH Fuel Clad and RCS **OR** Potential Loss of EITHER Fuel Clad or RCS **AND** Loss of ANY Additional Barrier	**FG1** Loss of ANY Two Barriers AND Potential Loss or Loss of Third Barrier

Operating Modes 1 through 4

Note: 1. At the Site Area Emergency level, there must be some ability to dynamically assess how far present conditions are from General Emergency.

2. The ability to escalate to higher emergency classes as an event degrades must be maintained. RCS leakage steadily increasing would represent an increasing risk to public health and safety.

Determination of Emergency Classification Level

Select values from the top of the columns on the next page, which describe specific Fission Product Barrier degradation. Select the higher value that applies from each barrier. Add the values to arrive at the total challenge to the Fission Product Barriers. The emergency classification is determined from the range of values shown in parentheses in the table above.

RECOGNITION CATEGORY F
FISSION PRODUCT BARRIER DEGRADATION
INITIATING CONDITION MATRIX

EAL	FUEL CLAD		RCS		CONTAINMENT	
	POTENTIAL LOSS (3)	**LOSS (4)**	**POTENTIAL LOSS (3)**	**LOSS (4)**	**POTENTIAL LOSS (1)**	**LOSS (2)**
1	CSF Core Cooling - Orange **OR** Heat Sink - Red[2]	CSF Core Cooling - Red	CSF RCS Integrity – Red **OR** Heat Sink - Red[2]	CSF Core Cooling - Yellow with subcooling < 0 °F	CSF Containment - Red **OR** Core Cooling - Orange > 15 min.	–
2	RCS Activity Failed Fuel Monitor, RT-8039, equal to or greater than 870 µCi/ml	RCS Activity Dose Equivalent Iodine greater than 300 µCi/gm	RCS Leak Rate Unisolable leak exceeding the capacity of one centrifugal charging pump in the normal charging mode.	RCS Leak Rate Leak rate greater than CVCS System's ability to maintain RCS inventory as indicated by loss of RCS subcooling.	Containment Pressure Greater than 6% hydrogen concentration in containment **OR** Containment pressure greater than 9.5 psig with neither containment spray nor RCFC running.	Containment Pressure Initial increase followed by rapid unexplained decrease **OR** Containment pressure or sump level not increasing as expected with LOCA conditions.
3	Core Exit Thermocouple ≥ 708°F	Core Exit Thermocouple 1200°F	SG Tube Rupture SG Tube has ruptured and the primary to secondary leak rate is greater than the capacity of one centrifugal charging pump.	SG Tube Rupture SG Tube is ruptured and has a non-isolable secondary steam release	–	SG Tube Leak Primary to secondary leakage greater than 150 gpd through any one steam generator with direct secondary side leakage to atmosphere
4	Reactor Vessel Water Level Plenum level less then 20%	–	–	–	Containment Bypass VALID increase in reading on area or ventilation monitors in areas adjacent to the containment boundary with a known LOCA inside containment.	Containment Isolation Containment isolation signal **AND** Valves not closed **AND** A pathway to the environment exists.
5	–	RCB Rad Monitor RT-8050 or RT-8051 greater than 100 R/hr **OR** Hatch Monitor greater than 222 mR/hr	–	RCB Rad Monitor RT-8050 or RT-8051 greater than 100 R/hr **OR** Hatch Monitor greater than 222 mR/hr	RCB Rad Monitor RT-8050 or RT-8051 greater than 1,000 R/hr **OR** Hatch Monitor greater than 2,222 mR/hr	–

Note: 1. The Fuel Clad barrier and the RCS barrier are weighted more heavily than the Containment Barrier. Unusual Event Initiating Conditions (ICs) associated with RCS and Fuel Clad barriers are addressed under SU6 and SU7.

2. CSF indicators must be valid; outside the immediate control of the operator.

STI: 33010739	**0ERP01-ZV-IN06**	**Rev. 6**	Page 1 of 7
Radiological Exposure Guidelines			
Quality	Non Safety-Related	Usage: **N/A**	Effective Date: 10/26/11

S. Korenek	N/A	N/A	Emergency Response Division
PREPARER	TECHNICAL	USER	COGNIZANT ORGANIZATION

Table of Contents Page

1.0 Purpose and Scope

1.1 This procedure provides guidance for authorizing radiological exposures in excess of 10CFR20 limits during response to emergency conditions at an Alert, Site Area Emergency, or General Emergency.

1.2 This procedure implements the requirements of the South Texas Project Electric Generating Station (STPEGS) Emergency Plan specific to radiological exposure guidelines during emergency conditions at an Alert, Site Area Emergency, or General Emergency.

2.0 Responsibilities

2.1 The Emergency Director is responsible for approving ALL requests for exposure extensions above 10CFR20 limits.

2.2 The following individuals are responsible for requesting exposure extensions, and tracking approvals for the following personnel:

2.2.1 Acting Radiological Manager - Onshift personnel until Technical Support Center activation.

2.2.2 Radiological Coordinator – Operations Support Center personnel

2.2.3 Radiological Manager – Technical Support Center, Security and Control Room personnel.

2.2.4 Radiological Director – Emergency Operations Facility and Offsite Field Teams.

3.0 Precautions and Limitations

3.1 Administrative dose limits are not applicable.

3.2 Emergency responders shall be authorized an exposure limit of 5 rem TEDE.

3.3 No individual shall knowingly exceed 10CFR20 exposure limits except when authorized to do so by the Emergency Director.

3.4 Upon Assembly and Accountability completion, ensure all personnel remaining in the Protected Area have Thermoluminescent Dosimetry, Document issue using Form 3, TLD Issuance Log.

3.5 Data Sheet 1, Emergency Exposure Approval, Data Sheet 2, Emergency Exposure Approval Log, shall be included in the individual dosimetry files which are forwarded to Records Management for retention. (10CFR20.2106) (ANI 80.1A)

4.0 References

4.1 STPEGS Emergency Plan

4.2 EPA 400-R-92-001, Manual of Protective Action Guides and Protective Actions for Nuclear Incidents

4.3 NUREG-0654 FEMA REP-1, Rev. 1, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants

4.4 0PGP05-ZV-0004, Emergency Plan Implementing Procedure Users Guide

5.0 Procedure

5.1 Verify individual(s) have been issued a Thermoluminescent Dosimeter.

5.2 Determine if alternative actions are available to preclude exposure extension. This should include the use of shielding, assignment of personnel with lower exposures, or rotation of personnel on the job.

5.3 To approve exceeding 10CFR20 exposure limits complete Data Sheet 1, Emergency Exposure Approval, for each individual.

5.4 Log individuals' names on Data Sheet 2, Emergency Exposure Approval Log, for all individuals approved to exceed exposure limits.

5.5 Brief individuals expected to receive exposures greater than 25 rem TEDE using Addendum 2, Risks Involved with Exposures Greater Than 25 rem TEDE.

6.0 Support Documents

6.1 Addendum 1, Emergency Cumulative Exposure Limits

6.2 Addendum 2, Risks Involved with Exposures Greater Than 25 rem TEDE

6.3 Data Sheet 1, Emergency Exposure Approval

6.4 Data Sheet 2, Emergency Exposure Approval Log

TEDE Exposure Limit	Required Approval to Exceed 10CFR20 Exposure Limit	Approval Documentation	Special Considerations
5-10 rem	Emergency Director	Data Sheet 1	• Approval to receive exposure in this range must be based on the protection of valuable property, life saving activities or protecting large populations.
10 - 25 rem	Emergency Director	Data Sheet 1	• Approval to receive exposure in this range must be based on life saving activities or protecting large populations.
>25 rem	Emergency Director	Data Sheet 1	• Approval to exceed this limit applies to life saving activities or protecting large populations. • 25 rem TEDE is planned exposure limit and should not be considered a maximum upper limit to removing personnel from life threatening environments. • The individual shall be a volunteer and be aware of the risks involved.

You have indicated that you are volunteering to be part of an emergency repair/damage control team. The nature of your task is such that you will probably receive an exposure to radiation that will be at a level above the normal limits. You need to have full awareness of the radiological risks involved. The purpose of this briefing is to make you aware of these risks.

The Emergency Director has authorized you to receive an emergency exposure. As you might recall from your training, our procedures allow such a once in a lifetime exposure. The emergency limits are based upon recommendations by the EPA.

There are two categories of risk associated with this type of radiological exposure that you should be fully aware of. These two risks are the immediate health effect and the delayed health effect.

IMMEDIATE HEALTH EFFECTS:

The immediate health effect of an acute exposure (a large dose within 24 hours) to radiation will vary with the individual and with the amount of exposure. Generally, as the exposure increases, the immediate health effect is more severe. The exposure you are projected to receive will most likely cause temporary blood changes which may make you temporarily more susceptible to illness. Also, although unlikely, you may experience temporary nausea, vomiting and diarrhea. Your level of exposure is well below the levels that have led to immediate fatalities. As a point of reference, at the 100 rem TEDE whole body dose, without medical treatment, approximately 1% of the exposed population would die. At an exposure of 450 rem TEDE, without medical treatment, 50% would die.

DELAYED HEALTH EFFECTS:

The delayed health effect of an acute exposure to radiation is an increase in the risk of premature death from cancer.

It is hard to estimate the increase in cancer risk; however, it is fair to state that your chance of getting cancer increases with every rem that you receive.

Statistics indicate that if a person 40 to 50 years of age were to receive 25 rem TEDE of exposure, then the risk of premature death due to cancer will increase by approximately 0.5%. Another way of looking at this would be to suppose what would happen if 1000 people were exposed to 25 rem TEDE. These same statistics indicate that we would expect approximately 5 of the 1000 individuals to die prematurely from cancer as a result of the 25 rem TEDE exposure (premature meaning an estimated 15 years of life lost).

In summary, if an individual 45 years old receives a 25 rem TEDE exposure, then it is possible that that person could die prematurely from cancer. The odds are roughly 1 in 200.

Data Sheet 1	Emergency Exposure Approval	Page 1 of 1

Name: ________________________________ SSN: _______________________________________

Date/Time: ____________________________ Discipline Group: ____________________________

Reason for Dose Extension Request: ____

____

Current exposure to date: ___________ Rem (TEDE) TLD No.: _____________________________

Exposure Authorized to: ____________________________ Rem (TEDE)

_____________________________________ _____________________________________

*Emergency Worker Emergency Director

* Required to exceed 10CFR20 TEDE exposure limit (5 rem).

Data Sheet 2	Emergency Exposure Approval Log	Page 1 of 1

Name	Approval to Exceed 10CFR20 Exposure limits		
	>5 rem TEDE	>10 - <25 rem TEDE	>25 rem TEDE

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
____ 6.0	***MAINTAIN* VCT Level – GREATER THAN 15% WITH CHARGING PUMP SUCTION ALIGNED TO VCT (CP004)** • Auto makeup • Manual makeup	PERFORM the following: **a.** TRIP the Reactor. **b.** INITIATE Safety Injection. **c.** GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.
____ 7.0	**CHECK Main Turbine In Service**	GO TO Step 13.0.
____ 8.0	**CHECK For One Of The Following:** • **Leakage From Any One SG Is GREATER THAN OR EQUAL TO 75 gpd <u>AND</u> Continues To Increase At GREATER THAN OR EQUAL TO 30 gpd/hr** **OR** • **Leakage From Any One SG Is GREATER THAN OR EQUAL TO 75 gpd <u>AND</u> Loss of Continuous Radiation Monitoring**	PERFORM the following: **a.** COMPARE SG Tube Leak Rates to values listed in Addendum 6. **b.** <u>IF</u> Mode 3 is REQUIRED by Addendum 6, <u>THEN</u> PERFORM the following: **1)** COMMENCE plant shut down per 0POP03-ZG-0006, Plant Shutdown From 100% To Hot Standby, per response time requirements of Addendum 6. **2)** GO TO Step 10.0. **c.** <u>IF</u> Shutdown <u>NOT</u> required, <u>THEN</u> RETURN to procedure and step in effect.
____ 9.0	**PERFORM The Following:** ____ **a. COMMENCE Plant Shutdown Per 0POP04-TM-0005, Fast Load Reduction Per The Response Time Requirements Of Addendum 6** ____ **b. CONTINUE performance of 0POP04-TM-0005, Fast Load Reduction until TURBINE TRIPPED**	

CAUTION

Post Reactor Shutdown conditions in the primary to secondary leakage (RCS temperature and pressure decreasing and SG pressure increasing) may reduce the SG Tube Leakage Rate. Plant Shutdown and Cooldown Rates should be based on the initial or increased leakage and **NOT** reduced leakage due to the Post Shutdown conditions.

Action Level	Leak Rate	Increasing Leak Rate	Response Times (1)
3	≥ 75 gpd (4)	Rate of inc ≥ 30 gpd/hr	Reduce Rx PWR to ≤ 50% in 1 hr AND Mode 3 in the next 2 hr
3	≥ 75 gpd AND Loss of Continuous Radiation Monitoring (3)	NA	Reduce Rx PWR to ≤ 50% in 1 hr AND Mode 3 in the next 2 hr
3	≥ 150 gpd	NA	Mode 3 ≤ 6 hr
2	≥ 75 gpd (4)	Rate of inc < 30 gpd/hr	Mode 3 ≤ 24 hr
1	< 75 gpd AND Loss of Continuous Radiation Monitoring (3)	N/A	Continued Operations (2)
1	≥ 30 gpd	N/A	Continued Operations (2)
Increased Monitoring	≥ 5 gpd	N/A	Continued Operations (2)

(1) Response times are the maximum times allowed. Power Reduction and Mode Change(s) may be completed in less time.

(2) Continued Operations per Plant Management direction. Refer to 0PGP03-ZO-0041, Action For Monitoring Primary to Secondary Leakage.

(3) Loss of Continuous Radiation Monitoring as defined in 0PGP03-ZO-0041, Action for Monitoring Primary to Secondary Leakage.

(4) With a continued increase in leakage rate over 30 minute time interval per the next column.

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Control Rod Malfunction			
Quality	**Usage: IN HAND**	**Enhanced Off-Normal Procedure**	
Safety-Related	**CONTROLLING STATION**	Effective Date: 08/28/2012	

D. Rohan	P. Travis	Crew 2E	Generation Support
PREPARER	TECHNICAL	USER	COGNIZANT DEPT

Table of Contents

This Procedure is Applicable in Modes 1, 2, and 3

PURPOSE

1. This procedure provides instructions to stabilize the plant and recover from the following types of rod malfunctions during operation in Mode 1 or 2:
   - Dropped or misaligned rod(s)
   - Immovable rod(s)
   - Spurious rod movement
   - Shutdown Rod not fully withdrawn
2. Provide instructions to recover dropped or misaligned rod(s) in Mode 3.
3. Provide instructions to recover from rod(s) which fail to fully insert following a Reactor Trip or plant shutdown.
4. The applicable portions of this procedure may be used to provide guidance in realigning rods that are less than 12 steps from the Group Step Counter Demand position at the discretion of the Shift Manager/Unit Supervisor.

SYMPTOMS OR ENTRY CONDITIONS

1. Dropped or Misaligned Rod
   a. A deviation of greater than 12 steps between Digital Rod Position Indication (DRPI) and Group Step Counter Demand position indications for any rod.
   b. A deviation between DRPI and Group Step Counter Demand position indications for any rod that requires realignment.
   c. Rod Bottom LED on DRPI LIT.
   d. Decreasing RCS Tavg.
   e. Any of the following annunciators LIT:
      - "PR UPPER DET FLUX DEV HI/AUTO DEF" Lampbox 5M03 Window A-3
      - "RPI TRBL" Lampbox 5M03 Window A-5
      - "PR LOWER DET FLUX DEV HI/AUTO DEF" Lampbox 5M03 Window B-3
      - "PR CHANNEL DEV" Lampbox 5M03 Window C-3
      - "AXIAL FLUX DIFFERENCE HI" Lampbox 5M03 Window D-3
      - "ROD SUPV MNTR ROD POSITION TRBL" Lampbox 5M03 Window D-5
      - "ROD BOTTOM" Lampbox 5M03 Window F-4

SYMPTOMS OR ENTRY CONDITIONS (Continued)

2. Immovable Rod
   a. Inability to insert and/or withdraw any control rod.
   b. Misaligned rod.
   c. "ROD CONT URGENT ALARM" LIT — Lampbox 5M03 Window B-5
3. Spurious Rod Movement
   a. Unexpected rod motion with "ROD BANK SELECTOR SW" in MANUAL or AUTO.
4. Shutdown Rod **NOT** Fully Withdrawn
   a. "ROD SUPV MNTR ROD POSITION TRBL" LIT — Lampbox 5M03 Window D-5
   b. Any shutdown rod observed to be **NOT** withdrawn to the full out rod position.
5. Rod(s) fail to fully insert following a Reactor Trip or shutdown as indicated by absence of rod bottom light(s) lit.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 1.0	**ENSURE "ROD BANK SEL" Switch In MANUAL {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**	

NOTE

Steps 1.0 through 3.0 are IMMEDIATE ACTION Steps.

_____ **4.0 CHECK For Misaligned Rods:**

_____ **a. CHECK All Rods – ANY RODS MISALIGNED**

a. GO TO Step 5.0.

_____ **b. CHECK All Rods – ONLY ONE ROD MISALIGNED**

b. IF two or more rods are **NOT** aligned in MODES 1 OR 2, THEN:

1) REFER TO Technical Specification 3.1.3.1 Action d for appropriate action.

2) 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.

3) 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

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
_____ 6.0	**CHECK Tavg – WITHIN 1.5°F OF Tref**	*MAINTAIN* Tavg within 1.5°F of Tref by any of the following methods: • ADJUST Turbine load • ADJUST RCS boron concentration • IF Turbine is offline, THEN ADJUST demand on the Steam Generator PORVs OR Steam Dumps.

NOTE

- An Urgent Failure in the Logic Cabinet prevents all automatic and manual rod motion in overlap.
- An Urgent Failure in a Power Cabinet prevents all rod motion by the rods powered from the failed cabinet.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 7.0	**CHECK "ROD CONT URGENT ALARM" Lampbox 5M03 Window B-5 – EXTINGUISHED**	DISPATCH an Operator to determine alarm status of rod control logic and power cabinets. {60 ft EAB Room 323}
_____ 8.0	**NOTIFY I&C Personnel To Assist In Troubleshooting And Diagnosis Of The Malfunctioned Rod(s)**	
_____ 9.0	**PERFORM Required Actions And Conditions Of Technical Specification 3.1.3.1**	

NOTE

Step 10.0 may be performed concurrently with the rest of this procedure.

_____ **10.0 NOTIFY The Following Of The Rod Malfunction:**

- _____ **Reactor Engineering Supervisor**
- _____ **Plant Operations Manager**

NOTE

Quadrant Power Tilt Ratio (QPTR) can be monitored on the Plant Computer by selecting the "FT: Radial Flux Tilts" display under Nuclear Application Programs.

_____ **11.0 *MONITOR* The Following At Least Once Every Hour:**

- Quadrant Power Tilt Ratio (QPTR)
- Axial Flux Difference (AFD)

_____ **12.0 CHECK QPTR And AFD – WITHIN LIMITS OF TECHNICAL SPECIFICATIONS**

- 3.2.1
- 3.2.4

RESPONSE NOT OBTAINED:

PERFORM the following:

a. *MONITOR* the length of time that the limits of Technical Specification 3.2.1 or 3.2.4 are exceeded.

b. REFER TO Technical Specifications 3.2.1 and 3.2.4 for appropriate action.

c. IF action statements of Technical Specifications 3.2.1 and 3.2.4 for power reduction can NOT be complied with, THEN PERFORM the following:

1) TRIP the Reactor.

2) GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

An immovable rod should **NOT** be considered untrippable unless I&C personnel have verified that the stationary gripper, movable gripper, and lift coils are being energized with the correct currents and in the proper sequence.

_____ **13.0 CHECK All Rods – TRIPPABLE**

PERFORM the following:

a. IF two or more rods are determined **NOT** trippable, THEN:

1) TRIP the Reactor.

2) GO TO 0POP05-EO-EO00, Reactor Trip Or Safety Injection.

b. IF only one rod is determined **NOT** trippable, THEN:

1) REFER TO Technical Specification 3.1.3.1 Action a for appropriate action.

2) 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 determining rod is **NOT** trippable.

_____ **14.0 CHECK DRPI – OPERABLE**

REFER TO Technical Specification 3.1.3.2 for appropriate action.

STEP	ACTIONS/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
_____ 15.0	**CHECK All Rods – ALIGNED**	PERFORM the following: **a.** IF two or more rods are **NOT** aligned in MODES 1 OR 2, THEN: **1)** REFER TO Technical Specification 3.1.3.1 Action d for appropriate action. **2)** 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. **b.** IF only one rod is **NOT** aligned OR in Mode 3, THEN GO TO Addendum 2, Recovery of Misaligned Rods. ---
_____ 16.0	**CHECK Rod Status – ALL MOVABLE**	IF any rod(s) is aligned AND **NOT** movable, THEN PERFORM the following: **a.** *MAINTAIN* compliance with Rod Insertion Limits and overlap specified in the Core Operating Limits Report. **b.** *MAINTAIN* affected bank(s) within 12 steps of the immovable rod(s). **c.** IF the immovable rod becomes misaligned, THEN GO TO Addendum 2, Recovery of Misaligned Rods. **d.** IF only one rod is immovable, THEN REFER TO Technical Specification 3.1.3.1 Action b for appropriate action. **e.** IF two or more rods are immovable, THEN REFER TO Technical Specification 3.1.3.1 Action c for appropriate action. ---

_____	**17.0**	**CHECK All Shutdown Rods – FULLY WITHDRAWN**	IF two or more Shutdown Rods are **NOT** fully withdrawn, THEN REFER TO Technical Specifications 3.1.3.5 and 3.0.3 for appropriate action.
			--
_____	**18.0**	**RECORD The Following In The Control Room Log:**	
_____		• **Core location of malfunctioned rod(s)**	
_____		• **Digital Rod Position Indication (DRPI) for malfunctioned rod(s)**	
_____		• **Affected bank(s)**	
_____		• **Group Step Counter Demand position of affected bank(s)**	
_____		• **Type of malfunction (e.g., misaligned, immovable, etc)**	
_____		• **Date and time malfunction occurred**	
			--

– END –