Exam Bank No.: 27 **Last used on an NRC exam:** 1999

Following a Loss of Offsite Power (LOOP), which ONE of the following statements describes the Pressurizer heater groups that will be available to maintain Pressurizer pressure?

A. Backup heater groups A and B only

B. Backup heater groups D and E only

C. All Backup heater groups

D. Control heater group C only

Answer: A Backup heater groups A and B only

Exam Bank No.: 27 **RO Outline Number:**

K/A Catalog Number: 011 K2.02 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.1 **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 the power supplies to the pressurizer heaters.

Reference: LOT201.14 Rev 9

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT - Backup heater groups A and B are supplied power from Class 1E Load Centers E1A1 and E1C1 respectively.

B: INCORRECT - Only backup heater groups A and B have Class 1E power supplies; Backup heater groups D and E have Non-Class power supplies.

C: INCORRECT - Backup heater groups D and E are supplied power from Non-Class Load Centers 1N and 1P respectively.

D: INCORRECT - Control heater group C is supplied power from Non-Class Load Center 1N

Question Level: F **Question Difficulty** 3

Justification:

The candidate must know which heater groups are safety related (i..e ESF diesel powered).

Exam Bank No.: 34 **Last used on an NRC exam:** 1999

Unit 1 is operating at 90% power when the following annunciator and Bistable Monitoring Panel (BSMP) indications are received:

- Annunciator 5M02, Window F-2, UV RX PRETRIP
- BSMP - RCP1B U/V RX TRIP

NO other alarms or changes in indications or equipment status are observed.

Which ONE of the following statements is correct?

A. Control of RCP 1B from the Control Room has been lost.

B. The UV device on Auxiliary Bus 1G should have tripped RCP 1B.

C. The Reactor Protection System failed to actuate and initiate a reactor trip.

D. The RCP 1B UV device has failed.

Answer: D The RCP 1B UV device has failed.

Exam Bank No.: 34 **RO Outline Number:**

K/A Catalog Number: 003 K3.04 **Tier:** 2 **Group/Category:** 1

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

Knowledge of the effect that a loss or malfunction of the Reactor Coolant Pump System (RCPS) will have on the following: Reactor Protection System (RPS)

STP Lesson: LOT 201.20 **Objective Number:** 3822

DESCRIBE the effect on the solid state protection system bistables status upon channel deenergization.

Reference: 0POP09-AN-5M2-F-2, Rev 12; 9-E-RC03-02, Rev 9; 9-E-RC01-01, Rev 10; 9-E-PC12-01, Rev 10; 9-Z-42111 Rev 13

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☑ **Attachment:** 0POP09-AN-5M2-F-2; 9-E-RC03-02; 9-Z-42111

Source: New **Modified from**

Distractor Justification

A: Incorrect; Cues given indication a RPS interface. Loss of capability to control the RCPs from the Control Room would indicated by loss of breaker (red/green) status indication; the question stem stated no change in status or operation of any equipment.

B: Incorrect; The UV trip input to the RCP control circuit is supplied by a UV device monitoring the 13.8 Kv Auxiliary Bus.

C: Incorrect; The reactor trip on RCP UV is 2/4 logic; only one input is present from cue given.

D: Correct; Cues given indicate the presence of an UV input to the RPS from RCP 1B; since no other conditions changed as stated in the stem, the input must be due to a failed sensor.

Question Level: H **Question Difficulty** 4

Justification:

The student must analyze the conditions given and use their knowledge of the SSPS inputs and logic and electrical protection circuitry to conclude that there has been a single component failure and that the indicated plant response to this failure is expected.

Exam Bank No.: 35 **Last used on an NRC exam:** 2007

Which one of the following automatic actions will occur as Instrument Air pressure decreases below 100 psig?

A. 1-IA-PV-9983, Instrument Air Dryer Bypass Valve will OPEN

B. 1-CV-FV-0011, Letdown Orifice Header Isolation Valve will CLOSE

C. 1-IA-PV-8568, Instrument Air to Yard Valve will CLOSE

D. 1-IA-PV-9785, Service Air Isolation Valve will CLOSE

Answer: D 1-IA-PV-9785, Service Air Isolation Valve will CLOSE

Exam Bank No.: 35 **RO Outline Number:**

K/A Catalog Number: 078 K4.02 **Tier:** 2 **Group/Category:** 1

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

Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Cross-over to other air systems.

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, R11, pages 2 and 3

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - 1-IA-PV-9983 does not automatically open until 80 psig decreasing.

B: INCORRECT - 1-CV-FV-0011 does not start to fail closed (on loss of IA) until 80 psig decreasing.

C: INCORRECT - 1-IA-PV-8568 does not automatically close until 90 psig decreasing.

D: CORRECT - 1-IA-PV-9785 automatically closes at 100 psig decreasing therefore, it would already be closed based on the given conditions.

Question Level: F **Question Difficulty** 2

Justification:

The applicant must apply the given conditions to his/her knowledge of the instrument air system setpoints and automatic actions to determine the current expected alignment of the system.

Exam Bank No.: 128 **Last used on an NRC exam:** 1999

Given the following:

- Unit 1 is in Mode 2 performing a Reactor startup.
- Turbine Impulse Pressure channel PT-506 fails high.

Which ONE of the following Reactor Trip signals is ENABLED due to the PT-506 failure?

Reactor Trip from:

A. Low Pressurizer Pressure.

B. Loss of Reactor Coolant Flow in 1 Loop.

C. Pressurizer Low Water Level.

D. AMSAC

Answer: A Low Pressurizer Pressure

Exam Bank No.: 128 **RO Outline Number:**

K/A Catalog Number: 045 K1.18 **Tier:** 2 **Group/Category:** 2

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

Knowledge of the physical connections ans/or cause-effect relationships between the MT/G system and the following: RPS.

STP Lesson: LOT 201.20 **Objective Number:** 3832

Describe the reactor protection system control and permissive interlocks including inputs, setpoints, coincidence, and functions

Reference: LOT201.20

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT - The failure of PT-506 will satisfy the logic for P-7 and re-instate the low power trips that are automatically blocked below 10% power.

B: INCORRECT - Loss of a single RC loop (low flow) reactor trip is enabled at > P-8 (40% NI power) .

C: INCORRECT - There is no low pressurizer level reactor trip.

D: INCORRECT - AMSAC is not a reactor trip

Question Level: H **Question Difficulty** 3

Justification:

The candidate must determine what effect the failure has on the RPS, then using his knowledge of permissives and reactor trips determine which trip is enabled.

Exam Bank No.: 150 **Last used on an NRC exam:** 2005

Which ONE of the following groups of setpoints and coincidences will cause a Steam Line Isolation Signal?

A. Steam line pressure less than 735 psig, 2/3 channels on 1/4 steam lines with Low Steamline Pressure SI NOT blocked.

B. Containment HIGH-II pressure greater than 5 psig on 1/3 channels.

C. Steam line pressure less than 735 psig, 2/3 channels on 2/4 steam lines with Low Steamline Pressure SI blocked.

D. Containment HIGH-I pressure greater than 2 psig on 2/3 channels.

Answer: A Steam line pressure less than 735 psig, 2/3 channels on 1/4 steam lines with Low Steamline Pressure SI NOT blocked.

Exam Bank No.: 150 **RO Outline Number:**

K/A Catalog Number: 013 K4.03 **Tier:** 2 **Group/Category:** 1

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

Knowledge of the ESFAS design feature(s) and /or interlock(s) which provide for the following: Main Steam Isolation System

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: LOT201.20 lesson handout #2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT: This set of conditions will result in a main steam line isolation.

B: INCORRECT: HIGH-II pressure must be sensed on 2/3 channels (setpoint is 3 psig).

C: INCORRECT: Setpoint is correct, but actuation will not occur if low steamline pressure SI is blocked.

D: INCORRECT: HIGH-I pressure results in a Safety Injection actuation.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze each situation with repect to the actuation criteria to determine is a Steam Line Isolation will occur.

Exam Bank No.: 725 **Last used on an NRC exam:**

Given the following:

- Unit 1 is performing a plant shutdown per 0POP03-ZG-0006, Plant Shutdown from 100%.
- P-13 TURB LOAD LESS THAN 10 PRCT is illuminated.
- P-10 MAN BLOCK INT/LO PR RX TRP PERM is extinguished.
- Due to an excessive EHC leak the Main Turbine was tripped.

Which ONE of the following correctly reflects the status of the plant?

P-7 POWER OPER RX TRIPS BLKD is....

A. extinguished and Reactor Trip Breakers are open.

B. extinguished and the Reactor is critical.

C. illuminated and the Reactor is critical.

D. is illuminated and Reactor Trip Breakers are open.

Answer: C illuminated and the Reactor is critical.

Exam Bank No.: 725 **RO Outline Number:**

K/A Catalog Number: 012 A3.02 **Tier:** 2 **Group/Category:** 1

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

Ability to monitor automatic operation of the RPS, including: Bistables

STP Lesson: **Objective Number:**

Reference: 0POP09-AN-05M24, Permissive Lampbox 5M24 Information (Rev 5)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: P-7 Light is illuminated <10%

B: INCORRECT: P-7 Light is illuminated <10%

C: CORRECT: P-7 Light is illuminated <10%

D: INCORRECT: Reactor will not trip on turbine trip when Power Range NI's are less than 50%. Power Range NI's are less than 10% since P-10 is extinguished

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine the status of permissive functions based on the given plant status.

Exam Bank No.: 751 **Last used on an NRC exam:**

Given the following:

- Unit 2 has experienced a Large Break LOCA
- The ECCS is being aligned for Cold Leg Recirculation
- The Control Room operators have performed all the required actions of 0POP05-EO-ES13, Transfer to Cold Leg Recirculation, and NO Containment Emergency Sump Outlet Valves have opened.

Which ONE of the following RWST levels is the minimum level at which ANY pump taking suction on the RWST should be secured per 0POP05-EO-EC11, Loss of Emergency Coolant Recirculation?

A. 16,500 gallons (3%)

B. 26,500 gallons (5%)

C. 32,500 gallons (6%)

D. 42,500 gallons (8%)

Answer: C 32,500 gallons (6%)

Exam Bank No.: 751 **RO Outline Number:**

K/A Catalog Number: W/E11 EK2.2 **Tier:** 1 **Group/Category:** 1

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

Loss of Emergency Coolant Recirculation, EK2.2 – Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-EC11, Loss of Emergency Coolant Recirculation (Rev 16)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Refer to Correct answer

B: INCORRECT: Refer to Correct answer

C: CORRECT: This is the procedurally required RWST level at which any pumps taking a suction on the RWST must be stopped.

D: INCORRECT: Refer to Correct answer

Question Level: F **Question Difficulty** 3

Justification:

Must know procedural criteria for stopping pumps taking a suction on the RWST during accident conditions.

Exam Bank No.: 929 **Last used on an NRC exam:**

Given the following conditions:

- Reactor power is at 15% with the SGWLC system in automatic on the Main Feed Reg Valves (MFRV's)
- #12 SGFPT is in service and in AUTO
- The SGFP Master Controller is in AUTO.
- Steam header pressure transmitter PT-557 fails high.

Which ONE of the following describes the INITIAL plant response?

	SGFP speed	Main Feed Reg Valves
A.	DECREASES	OPEN
B.	DECREASES	CLOSE
C.	INCREASES	OPEN
D.	INCREASES	CLOSE

Answer: D INCREASES□CLOSE

Exam Bank No.: 929 **RO Outline Number:**

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 judgments based on operating characteristics, reactor behavior, and instrument interpretation.

STP Lesson: LOT 202.15 **Objective Number:**

Reference: LOT 202.15, Steam Generator Water Level Control System (Rev 9)
LOT 202.09, Steam Dumps (Rev 12)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Steam dumps would open (not close) to lower false steam pressure signal due to PT-557 failing high

B: INCORRECT: SGFP speed would increase (not remain unchanged) to re-establish programmed feed/steam D/P due to PT-557 failing high.

C: INCORRECT: Same as answer A

D: CORRECT: Steam dumps would open (not close) to lower false steam pressure signal due to PT-557 failing high. SGFP speed would increase to re-establish feed/steam programmed D/P. Low Power FRV's would initially close in response to SG swell induced from the steam dumps opening

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine how the PT-557 failure will affect 3 control systems based on the given plant conditions.

Exam Bank No.: 1053 **Last used on an NRC exam:** 2001

Which ONE of the following statements is correct concerning the Class 1E 125 VDC electrical power?

A. The loss of all battery chargers for any Train will cause that corresponding battery to supply the DC Bus loads for a minimum of 12 hours.

B. The loss of a single battery charger for any Train will cause the standby charger to automatically connect to the corresponding bus to supply the DC Bus loads.

C. On a loss of all offsite AC, the battery chargers fail and the DC Buses are supplied by their battery until the battery chargers are manually returned to service following start of the ESF DGs.

D. On a loss of all offsite AC, battery chargers lose power, the ESF DGs start and the 1E load centers sequence on, then the chargers resume powering the DC buses.

Answer: D On a loss of all offsite AC, battery chargers lose power, the ESF DGs start and the 1E load centers sequence on, then the chargers resume powering the DC buses.

Exam Bank No.: 1053 **RO Outline Number:**

K/A Catalog Number: APE 058 AK1.01 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.8 **10CFR Reference:** 55.41(b)(8)

Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: Battery charger equipment and instrumentation

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

Reference: POP02-EE-0001, LOT 201.37, Class 1E 125 VDC System (Rev 8)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Batteriesare rated for 2 hrs. minimum, not 12.

B: INCORRECT: The battery chargers do not have to be manually loaded or re-energized once AC power is restored.

C: CORRECT: Vital DC system responds as described.

D: INCORRECT: Standby chargers do not automatically connect to the DC buses.

Question Level: F **Question Difficulty** 3

Justification:

Must know how the Vital DC distribution system reacts upon a loss of AC power to the site.

Exam Bank No.: 1101 **Last used on an NRC exam:**

In accordance with 0POP08-FH-0009, Core Refueling, which of the below is an assigned task for a Licensed Control Room Operator during refueling operations?

A. Inform the Core Load Supervisor of the next core location to have a fuel assembly loaded.

B. Operate the remote television monitoring equipment used to observe refueling activities.

C. Monitor the Core Monitoring NI channels during and following insertion of each fuel assembly.

D. Inform the Refueling Machine Operator when he/she can disengage from a newly seated fuel assembly.

Answer: C Monitor the Core Monitoring NI channels during and following insertion of each fuel assembly.

Exam Bank No.: 1101 **RO Outline Number:**

K/A Catalog Number: G2.1.44 **Tier:** 3 **Group/Category:** 1

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

Knowledge of RO duties in the control room during fuel handling, such as responding to alarms from the fuel handling area, communication with the fuel storage facility, systems operated from the control room in support of fueling operations, and supporting instrumentation

STP Lesson: **Objective Number:**

Reference: 0POP08-FH-0009, Core Refueling

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Not specified as duty for a Licensed Opeator in 0POP08-FH-0009, Core Refueling

B: INCORRECT: Not specified as duty for a Licensed Opeator in 0POP08-FH-0009, Core Refueling

C: CORRECT: per 0POP08-FH-0009, Core Refueling

D: INCORRECT: Not specified as duty for a Licensed Opeator in 0POP08-FH-0009, Core Refueling

Question Level: F **Question Difficulty** 3

Justification:

Must know the assigned responsibilities for Licensed Operators during refueling.

Exam Bank No.: 1125 **Last used on an NRC exam:**

Given the following:

- Unit 2 is in Mode 3 at normal operating temperature and pressure.
- RCB temperature is approximately 80 degrees F.
- An error during surveillance testing results in a spurious Safety Injection initiation.
- RCB temperature has risen to 108 degrees F.

Which one of the following correctly describes the RCB temperature response?

RCB temperature is:

A. below the Tech Spec limit, and is rising because there is no cooling water flow through the RCFC's.

B. below the Tech Spec limit, and is rising because RCFC cooling flow has transferred to CCW.

C. above the Tech Spec limit, and is rising because there is no cooling water flow through the RCFC's.

D. above the Tech Spec limit, and is rising because RCFC cooling flow has transferred to CCW.

Answer: B below the Tech Spec limit, and is rising because RCFC cooling flow has transferred to CCW.

Exam Bank No.: 1125 **RO Outline Number:**

K/A Catalog Number: 022 A1.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(5)

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

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

Reference: LOT 202.33, Reactor Containment Building HVAC (Rev 6)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: RCB temperature is below the TS limit howver, there is CCW cooling flow to the RCFC's.

B: CORRECT: RCB temperature is below the TS limit and there is cooling flow from the CCW system under these plant conditions.

C: INCORRECT: RCB temperature is below the TS limit.

D: INCORRECT: RCB temperature is below the TS limit.

Question Level: H **Question Difficulty** 3

Justification:

requires the candidate to evaluate RCB temperature against TS limits and to determine why the temperature is rising based on the conditions of the stem.

Exam Bank No.: 1340 **Last used on an NRC exam:** 2005

In accordance with 0PGP03-ZA-0010, Performing and Verifying Station Activities, which of the following methods are an approved way to assure that a working copy of a procedure is current?

1. Document/Records Management Tracking System.
2. Comparison to a Level 1 Station Controlled hardcopy procedure.
3. Review of daily listing of procedure changes for continual use operational procedures.
4. Verify the working copy is the same revision and contains the same Field Changes as the last completed copy of the procedure.

A. 1, 2, 3

B. 1, 2, 4

C. 1, 3, 4

D. 2, 3, 4

Answer: A 1, 2, 3

Exam Bank No.: 1340 **RO Outline Number:**

K/A Catalog Number: G2.1.21 **Tier:** 3 **Group/Category:**

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

Ability to verify the controlled procedure copy.

STP Lesson: LOT 507.01 **Objective Number:** 92186

Given the title of an administrative procedure, DISCUSS the requirements associated with the referenced procedure.

Reference: 0PGP03-ZA-0010, step 5.1.2 (Rev 27)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: CORRECT: Listed choices all agree with currrent station procedures.

B: INCORRECT: Choice #4 will verify that the procedure is the same as the last time it was completed, but will not address any changes that may have occurred since then.

C: INCORRECT: Choice #4 will verify that the procedure is the same as the last time it was completed, but will not address any changes that may have occurred since then.

D: INCORRECT: Choice #4 will verify that the procedure is the same as the last time it was completed, but will not address any changes that may have occurred since then.

Question Level: F **Question Difficulty** 2

Justification:

The candidate requires a knowledge of station procedures for verifying a procedure is current.

Exam Bank No.: 1359 **Last used on an NRC exam:**

Ten minutes after an accident, containment pressure indicates the following:

- PT-934 = 9.5 psig
- PT-935 = 9.6 psig
- PT-936 = 9.3 psig
- PT-937 = 9.4 psig

Which of the following describes the response of the Containment Spray System assuming bistables actuate at their exact setpoint and no ESF systems have been reset?

A. Containment Spray has not automatically actuated.

B Containment Spray has automatically actuated, but only the pumps have started.

C. Containment Spray has automatically actuated, but only the pump discharge valves have opened.

D. Containment Spray has automatically actuated. The pumps have started and their discharge valves have opened.

Answer: D Containment Spray has automatically actuated. The pumps have started and their discharge valves have opened.

Exam Bank No.: 1359 **RO Outline Number:**

K/A Catalog Number: 026 A3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.3 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the CSS, including: Pump starts and correct MOV positioning.

STP Lesson: LOT 201.11 **Objective Number:** 2009

GIVEN a plant or system condition, PREDICT the operation of the Containment Spray System

Reference: LOT201.20 lesson handout #2; LOT201.11 lesson handout, page 8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: Containment Spray actuates on a 2 of 4 logic at 9.5 psig.

B: INCORRECT: The valves directly open from the actuation signal

C: INCORRECT: The pumps will start providing there is still a Sequencer signal present. The stem of the question says that no ESF systems have been reset so the pumps will start also

D: CORRECT: Containment Spray actuates on a 2 of 4 logic at 9.5 psig. The valves directly open from the actuation signal. The pumps will start providing there is still a Sequencer signal present. The stem of the question says that no ESF systems have been reset so the pumps will start also.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the conditions required to actuate both the spray pumps and discharge valves. This knowledge must then be applied to the conditions given to determine the correct response.

Exam Bank No.: 1398 **Last used on an NRC exam:** 2007

Unit 2 was operating at 100% power when a LOCA outside Containment occurred. The following conditions exist:

- The crew is performing 0POP05-EO-EC12, LOCA Outside Containment.
- FHB Area Rad Monitors are alarming.
- Train ‘A' ECCS and Containment Spray have been isolated based on high FHB SI/CS sump level.
- RCS temperature is slowly rising.
- RCS Pressure continues to lower.

Based on these conditions, which one of the below correctly describes the plant status and action path for the crew?

A. The leak IS ISOLATED. Transition to another EOP to cool down and depressurize the plant to Cold Shutdown conditions.

B. The leak IS ISOLATED. Transition to another EOP to terminate Safety Injection before initiating plant cooldown.

C. The leak is NOT ISOLATED. Remain in 0POP05-EO-EC12, and sequentially isolate the remaining ECCS and CS trains in an attempt to stop the leakage.

D. The leak is NOT ISOLATED. Remain in 0POP05-EO-EC12, and begin isolating systems that connect with the RCS that are located in the MAB.

Answer: C The leak is NOT isolated. Remain in 0POP05-EO-EC12, and sequentially isolate the remaining ECCS and CS trains in an attempt to stop the leakage.

Exam Bank No.: 1398 **RO Outline Number:**

K/A Catalog Number: EPE 011 2.4.6 **Tier:** 1 **Group/Category:** 1

RO Importance: 3,7 **10CFR Reference:** 55.41(b)(10)

Large Break LOCA, 2.4.6 – Knowledge of EOP mitigation strategies

STP Lesson: LOT 504.46 **Objective Number:** 82656

From memory, STATE/IDENTIFY the method used to identify and isolate the leak in accordance with POP05-EO-EC12.

Reference: 0POP05-EO-EC12, R8, Steps 3 and 4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT - The leak is not isolated because RCS pressure is still lowering.

B: INCORRECT -The leak is not isolated because RCS pressure is still lowering.

C: CORRECT - In accordance with Step 3, if RCS pressure continues to lower after the first SI Train is isolated, contine with step 4 to sequentially isolate other trains.

D: INCORRECT - Based on indications given (FHB radiation monitors alarming), the leak is probably in the FHB. Isolating systems connecting to the MAB will not be done until FHB leakage sources have been isolated.

Question Level: H **Question Difficulty** 3

Justification:

The applicant must first determine that the leak is not isolated (based on RCS pressure response) and then based on their knowledge of the procedure determine the correct action to take.

Exam Bank No.: 1401 **Last used on an NRC exam:**

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 one of the following correctly describes the effect of the valve failure?

RHR Train 'A' is.....

A. AVAILABLE to provide shutdown cooling since emergency instrument air 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 VAILABLE 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 **RO Outline Number:**

K/A Catalog Number: APE 024 G2.2.37 **Tier:** 1 **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: New **Modified from**

Distractor Justification

A: INCORRECT - Emergency IA tanks do not exist for these valves.

B: CORRECT - The valves fail as indicated, thus providing full cooling flow to the RCS.

C: INCORRECT - Bypass valve fails closed and the outlet valve fails open.

D: INCORRECT - The recirculation valve is a motor operated valve which would not be affected by the loss of air.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze the given conditions to determine the affect on the system and it availability.

Exam Bank No.: 1568 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in a reduced inventory condition.
- RCP Seal Injection has been secured.
- SG primary side manways have been removed.
- All RHR cooling is lost.
- Containment sump levels are rising.

The loss of RHR cooling has resulted in rising sump levels due to ...

A. rising RCS density resulting in leakage out of the RCP seals.

B. lowering RCS density resulting in leakage out of the RCP seals.

C. rising RCS density resulting in leakage from open SG primary side manways.

D. lowering RCS density resulting in leakage from open SG primary side manways.

Answer: D lowering RCS density resulting in leakage from open SG primary side manways.

Exam Bank No.: 1568 **RO Outline Number:**

K/A Catalog Number: APE 025 AA1.11 **Tier:** 1 **Group/Category:** 1

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(5)

Loss of RHR System, Ability to operate and/or monitor the following as they apply to the Loss of Residual Heat Removal System: Reactor Building sump level indicators

STP Lesson: **Objective Number:**

Reference: 0POP04-RH-0001, LOT 204.01

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified from**

Distractor Justification

A: INCORRECT: The loss of RHR will result in a heatup, but density will decrease, not increase.

B: INCORRECT: The loss of RHR will result in a heatup with a corresponding decrease in density causing water level to rise. Even with seal injection secured under these conditions, water will not flow out of the seals when the water level rises.

C: INCORRECT: The loss of RHR will result in a heatup, but density will decrease, not increase

D: CORRECT: A loss of RHR will result in a heat-up and decrease in density of the RCS. The expanding RCS fluid could then spill out open manways and end up in the containment building sumps.

Question Level: H **Question Difficulty** 3

Justification:

Requires candidate to determine the effects of a loss of RHR under the given plant conditions.

Exam Bank No.: 1589 **Last used on an NRC exam:**

If high radiation is detected by an Area Radiation Monitor detector, the alarm will be displayed on the Radiation Monitoring System (RMS) operator console (RM11) located in the Control Room.

Which one of the following correctly lists ALL of the RM11 operator console locations where the alarm will be displayed OUTSIDE the Control Room?

A. Emergency Operations Facility

B. Technical Support Center

C. Technical Support Center
AND
Health Physics Office

D. Emergency Operations Facility
AND
Health Physics Office

Answer: C Technical Support Center
AND
Health Physics Office

Exam Bank No.: 1589 **RO Outline Number:**

K/A Catalog Number: APE 061 AK2.01 **Tier:** 1 **Group/Category:** 2

RO Importance: 2.5 **10CFR Reference:** 55.41(b)(7)

Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: Detectors at each ARM system location.

STP Lesson: LOT 202.41 **Objective Number:** 92124

LIST the types of alarms associated with the Area Radiation Monitoring System (ARMS)

Reference: LOT202.41 lesson plan page 42

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT - there is not a console in the Emergency Operations Facility
B: INCORRECT - incomplete listing
C: CORRECT - correctly lists all locations
D: INCORRECT - an operator console is not located in the Emergency Operations Facility

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the radiation monitoring system equipment locations.

Exam Bank No.: 1624 **Last used on an NRC exam:**

During the performance of POP04-RC-0001, High Reactor Coolant System Activity, the Unit Supervisor directs you to check RT-8039, Failed Fuel Monitor, greater than the ALERT setpoint.

Which ONE of the following correctly describes how this can be accomplished?

A. On the QDPS Qual PAMS display, verify the RT-8039 readout is flashing.

B. On the QDPS Qual PAMS display, verify the RT-8039 readout is backlit.

C. On the RM-11 CRT display, verify RT-8039 indicates yellow or red.

D. On the RM-11 CRT display, verify RT-8039 indicates magenta.

Answer: C On the RM-11 CRT display, verify RT-8039 indicates yellow or red.

Exam Bank No.: 1624 **RO Outline Number:**

K/A Catalog Number: APE 076 AA2.04 **Tier:** 1 **Group/Category:** 2

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(11)

Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: Process effluent radiation chart recorder.

STP Lesson: LOT 202.41 **Objective Number:** 68793

DESCRIBE the meanings of the colors on the RM-11 display

Reference: LOT202.41 lesson plan page 29

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Flashing is not the method of alarm indication used on QDPS and RT-8039 indication is not found on the Qual PAMS display.

B: INCORRECT: While backlighting is the method to indicate an alarm on QDPS, RT-8039 indication is not found on the Qual PAMS display.

C: CORRECT: RT-3039 status is displayed on the RM-11and yellow/red indicates alert/high alarms respectively.

D: INCORRECT: Magenta is an indication of a communications failure and not related to radiation level.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have knowledge of the readout location for this monitor and be able to interpret the meaning of the indication.

Exam Bank No.: 1640 **Last used on an NRC exam:**

A Large Break Loss of Coolant Accident has occurred. The crew is presently performing the actions of POP05-EO-EO10, Loss of Reactor or Secondary Coolant.

Given the attached Qual PAMS indications and Containment Critical Safety Function Status Tree, which ONE of the following correctly describes the necessary procedure flowpath?

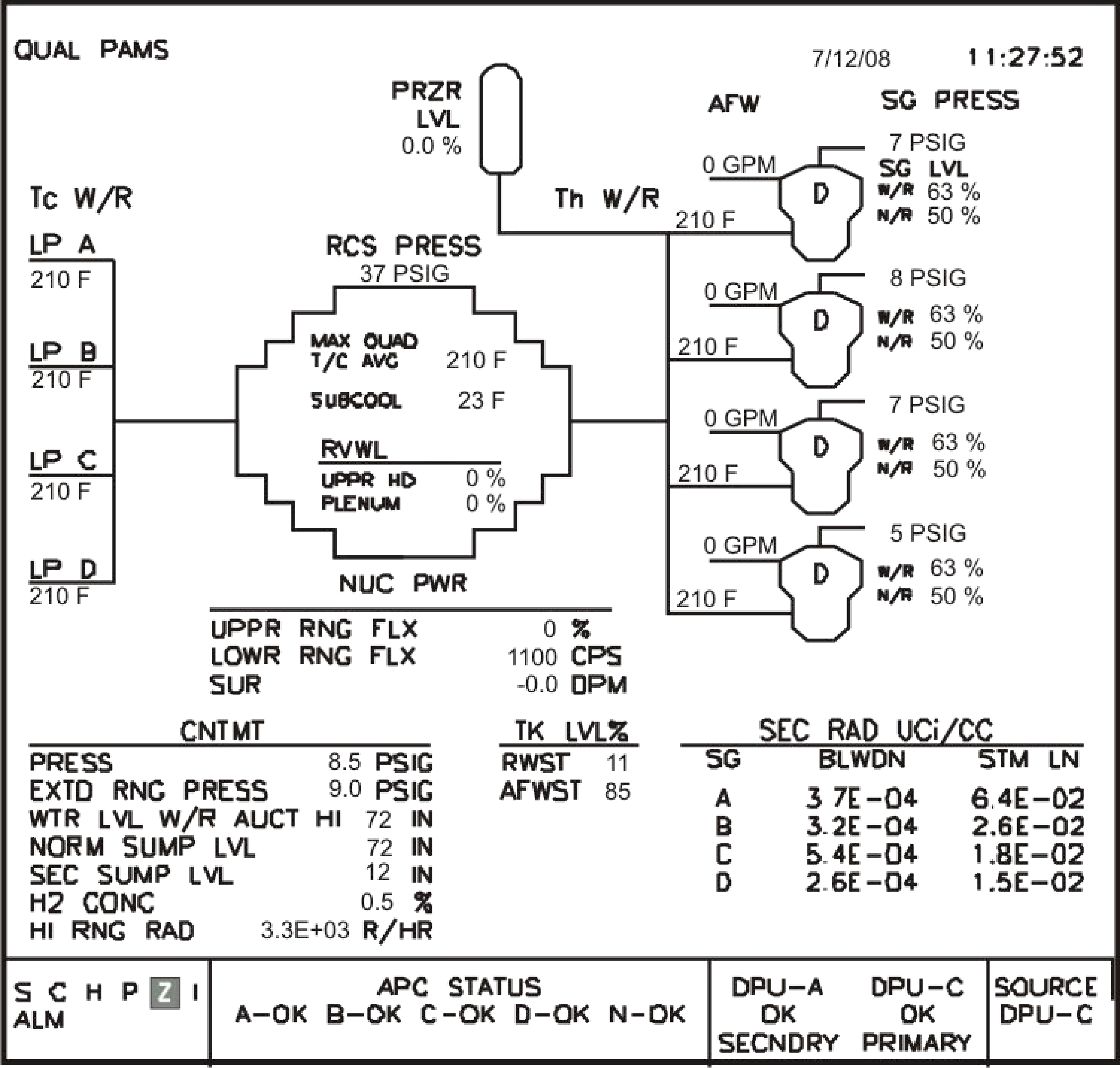
A. Continue performing the actions of POP05-EO-EO10, Loss of Reactor or Secondary Coolant.

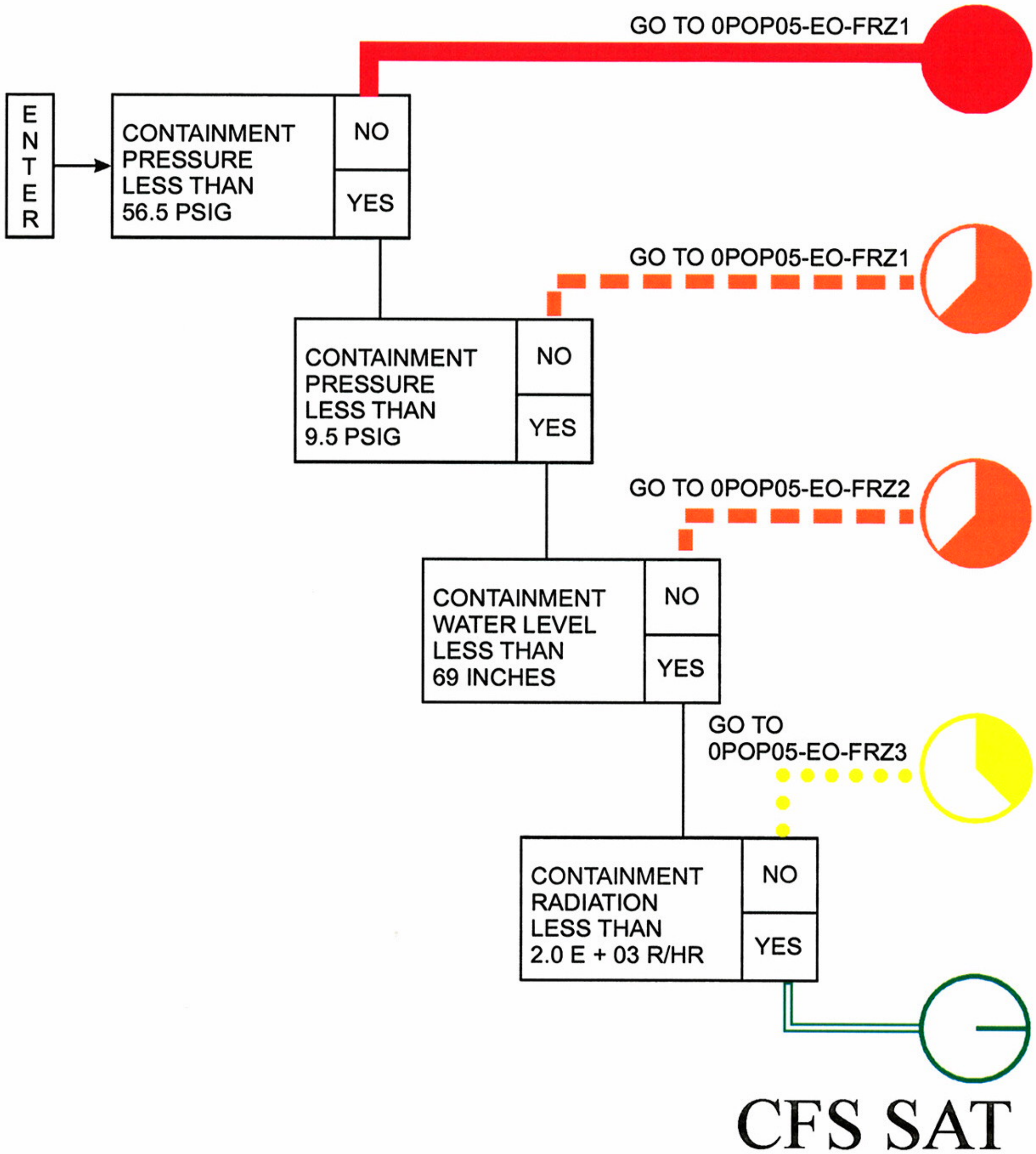
B. Transition to POP05-EO-FRZ1, Response to High Containment Pressure.

C. Transition to POP05-EO-FRZ2, Response to Containment Flooding.

D. Transition to POP05-EP-FRZ3, Response to High Containment Radiation Level.

Answer: C Transition to POP05-EO-FRZ2, Response to Containment Flooding


QUAL PAMS
7/12/08
11:27:52
PRZR
LVL
0.0 %
AFW
SG PRESS
7 PSIG
0 GPM
SG LVL
W/R 63 %
N/R 50 %
D
Tc W/R
Th W/R
210 F
LP A
210 F
RCS PRESS
37 PSIG
8 PSIG
0 GPM
W/R 63 %
N/R 50 %
210 F
MAX QUAD
T/C AVG 210 F
LP B
210 F
SUBCOOL 23 F
7 PSIG
0 GPM
W/R 63 %
N/R 50 %
RVWL
LP C
210 F
UPPR HD 0 %
PLENUM 0 %
210 F
5 PSIG
0 GPM
W/R 63 %
N/R 50 %
LP D
210 F
NUC PWR
210 F
UPPR RNG FLX 0 %
LOWR RNG FLX 1100 CPS
SUR -0.0 DPM
CNTMT
PRESS 8.5 PSIG
EXTD RNG PRESS 9.0 PSIG
WTR LVL W/R AUCT HI 72 IN
NORM SUMP LVL 72 IN
SEC SUMP LVL 12 IN
H2 CONC 0.5 %
HI RNG RAD 3.3E+03 R/HR
TK LVL%
RWST 11
AFWST 85
SEC RAD UCi/CC
SG BLWDN STM LN
A 3.7E-04 6.4E-02
B 3.2E-04 2.6E-02
C 5.4E-04 1.8E-02
D 2.6E-04 1.5E-02
S C H P Z I
ALM
APC STATUS
A-OK B-OK C-OK D-OK N-OK
DPU-A
OK
SECNDRY
DPU-C
OK
PRIMARY
SOURCE
DPU-C


GO TO 0POP05-EO-FRZ1
ENTER
CONTAINMENT PRESSURE LESS THAN 56.5 PSIG
NO
YES
GO TO 0POP05-EO-FRZ1
CONTAINMENT PRESSURE LESS THAN 9.5 PSIG
NO
YES
GO TO 0POP05-EO-FRZ2
CONTAINMENT WATER LEVEL LESS THAN 69 INCHES
NO
YES
GO TO 0POP05-EO-FRZ3
CONTAINMENT RADIATION LESS THAN 2.0 E + 03 R/HR
NO
YES
CFS SAT

Exam Bank No.: 1640 **RO Outline Number:**

K/A Catalog Number: EPE E15 EA2.1 **Tier:** 1 **Group/Category:** 2

RO Importance: 2.7 **10CFR Reference:** 55.41(b)(10)

Ability to determine and interpret the following as they apply to Containment Flooding: Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

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: POP05-EO-FO05, Containment CSF (Rev 1)

Attached Reference ☑ **Attachment:** Qual PAMS, POP05-EO-FO05

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: The CSF is not satisfied, so continuing in POP05-EO-EO10 is not the correct action to take.

B: INCORRECT: Containment pressure is less than 56.5 and 9.5 psig (red and orange path respectively).

C: CORRECT: Containment water level is greater than 69 inches

D: INCORRECT: Although the given radiation level exceeds the yellow path criteria, a higher priority orange path condition exists that must be addressed first.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must interpret the parameter display provided against the CSF Status Tree to determine the appropriate procedure for the plant conditions.

Exam Bank No.: 1653 **Last used on an NRC exam:**

Given the following plant conditions:

- Reactor power is at 45% with power ascension in progress following a refueling outage.
- All control systems are in automatic.
- A Pressurizer backup heater group has been manually energized to force spray flow.

The breaker for the backup heater group that was just energized trips open.

Assuming no Operator actions are taken, which ONE of the following describes a possible consequence of the breaker trip as the power ascension continues?

A. If a large Reactor Coolant System (RCS) cooldown occurs, the RCS boron concentration would lower.

B. If a large Reactor Coolant System (RCS) heatup occurs, the RCS boron concentration would lower.

C. If a large Reactor Coolant System (RCS) heatup occurs, the RCS boron concentration would rise.

D. If a large Reactor Coolant System (RCS) cooldown occurs, the RCS boron concentration would rise.

Answer: D If a large Reactor Coolant System (RCS) cooldown occurs, the RCS boron concentration would rise.

Exam Bank No.: 1653 **RO Outline Number:**

K/A Catalog Number: 004 K6.01 **Tier:** 2 **Group/Category:** 1

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

Knowledge of the effect of a loss or malfunction on the following CVCS components: Spray/heater combination in PZR to assure uniform boron concentration.

STP Lesson: LOT 506.01 **Objective Number:** 92158

In regards to the referenced procedure, DISCUSS the following: 1. Purpose and Scope, 2. Precautions, 3. Notes and Cautions

Reference: POP03-ZG-0005, step 4.34 (Rev 57)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Boron would be higher in the PZR, so outflow caused by the cooldown should result in RCS boron rising.

B: INCORRECT: An RCS heatup will result in inflow from the RCS into the PZR, thus RCS boron would not change.

C: INCORRECT: An RCS heatup will result in inflow from the RCS into the PZR, thus RCS boron would not change.

D: CORRECT: Less spray flow would result in PZR boron being higher than RCS. An RCS cooldown would result in outflow from the PZR, raising RCS boron concentration.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must understand the effect spray flow has on RCS/PZR boron differential and apply that knolwedge to an RCS heatup/cooldown.

Exam Bank No.: 1723 **Last used on an NRC exam:**

Per 0POP04-RA-0001, Radiation Monitoring System Alarm Response, which one of the following correctly matches a radiation monitor with its automatic action?

	Monitor	Automatic Action
A.	RT-8011, RCB Atmosphere	Closes Containment Supplemental Purge Isolation Valves.
B.	RT-8041, TGB Drain Monitor	Trips TGB Sump Number 1 Sump Pumps.
C.	RT-8033, EAB Intake	Trips Control Room Envelope Supply and Return Fans.
D.	RT-8039, Failed Fuel Monitor	Closes Letdown Isolation Valves LCV-0465 and LVC-0468.

Answer: B RT-8041, TGB Drain Monitor; Trips TGB Sump Number 1 Sump Pumps.

Exam Bank No.: 1723 **RO Outline Number:**

K/A Catalog Number: 068 K4.01 **Tier:** 2 **Group/Category:** 2

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

Knowledge of the design feature(s) and/or interlock(s) which provide for the following: Safety and environmental precautions for handling hot, acidic, and radioactive liquids.

STP Lesson: LOT 505.01 **Objective Number:** 92107

DISCUSS automatic actions expected to occur on entry conditions for the reference procedure.

Reference: POP04-RA-0001 (Rev 23), pages 14, 30, 39 and 41

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Although RT-8011 samples the containment atmosphere which is then released through supplemetary purge, this monitor provides indication only.

B: CORRECT: A high alarm on RT-8041 will stop the TGB Sump #1 pumps.

C: INCORRECT: RT-8033 will place filters in service to remove radioactive material, not trip the fans.

D: INCORRECT: Although RT-8039 monitors the letdown flow stream, it provides indication only.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have an understanding on the protection afforded by certain radiation monitors.

Exam Bank No.: 1724 **Last used on an NRC exam:**

Unit 1 is operating at 100% power with all systems in their normal lineup. An RCS leak develops in the Letdown Heat Exchanger room with the following Control Room indications:

- Pressurizer level indicates 57% and stable
- RCS pressure indicates 2235 psig and stable
- VCT level is 35 % and lowering
- LETDN HX OUTLET PRESSURE PI-0135 indicates 360 psig and stable
- LETDOWN HX OUTLET FLOW FI-0132 indicates 70 gpm and stable

Considering these indications, which ONE of the following AUTOMATIC actions has taken place?

A. TCV-0143, Letdown Temperature Divert Valve, positioned to the VCT

B. MOV-0468, Letdown Isolation Valve closed

C. FV-0011, Letdown Header Orifice Isolation Valve closed

D. PCV-0135, Letdown Pressure Control Valve throttled in close direction

Answer: D PCV-0135, Letdown Pressure Control Valve throttled in close direction

Exam Bank No.: 1724 **RO Outline Number:**

K/A Catalog Number: EPE E04 EK2.1 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.5 **10CFR Reference:** 55.41(b)(7)

Knowledge of the interrelations between the LOCA Outside Containment and the following: components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

STP Lesson: **Objective Number:**

Reference: LOT201.06.HO.01, Chemical and Volume Control System (Rev 13)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: TCV-0143 will automatically swap to the VCT on high L/D temperature only.

B: INCORRECT: MOV-0468 will automatically close on low PZR level (17%) and if closed, the letdown flow and pressure would be zero.

C: INCORRECT: FV-0011 will automatically cloe on low PZR level(17%) and if closed the letdown flow and pressure would be zero.

D: CORRECT: A leak in the letdown hx room will lower L/D pressure(PT-0135) throttling PCV-0135 to close to raise pressure to setpoint(approx. 360 psig). This will lower the indicated L/D flow(FI-0132). The combination of the FI-0132 and the leak will be matched by charging flow control FCV-0205 and PZR level will be returned to setpoint but VCT level will be lowering.

Question Level: H **Question Difficulty** 3

Justification:

Must determine what automatic actions have taken place by analyzing the given conditions.

Exam Bank No.: 1739 **Last used on an NRC exam:**

Given the following plant conditions:

- The plant is in Mode 4 cooling down to Mode 5
- All six RHR Pump suction MOVs have been energized
- The crew is preparing to place RHR in service
- Sensing line blockage has caused RCS wide range pressure transmitter PT-407 to stabilize at 700 psig
- RCS pressure has been verified to be 330 psig

Which ONE of the following correctly describes the effect of the blockage AND a method to mitigate the condition?

A. NONE of the RHR suction MOVs will open normally. The MOVs can be opened by holding the control switch in the OPEN position, and then opening the supply breaker once the valve is open in accordance with “skill of the craft” guidance in the Conduct of Operations Manual.

B. NONE of the RHR suction MOVs will open normally. The MOVs can be opened by temporarily transferring control to the Auxiliary Shutdown Panel in accordance with POP02-RH-0001, Residual Heat Removal System Operation .

C. ONLY TWO of the RHR suction MOVs will NOT open normally. The MOVs can be opened by holding the control switch in the OPEN position, and then opening the supply breaker once the valve is open in accordance with “skill of the craft” guidance in the Conduct of Operations Manual.

D. ONLY TWO of the RHR suction MOVs will NOT open normally. The MOVs can be opened by temporarily transferring control to the Auxiliary Shutdown Panel in accordance with POP02-RH-0001, Residual Heat Removal System Operation.

Answer: D ONLY TWO of the RHR suction MOVs will NOT open normally. The MOVs can be opened by temporarily transferring control to the Auxiliary Shutdown Panel in accordance with POP02-RH-0001, Residual Heat Removal System Operation.

Exam Bank No.: 1739 **RO Outline Number:**

K/A Catalog Number: 005 A2.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.7 **10CFR Reference:** 55.41(b)(7)

Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure modes for pressure, flow, pump motor amps, motor temperature, and tank level instrumentation.

STP Lesson: LOT 201.09 **Objective Number:** 4245

GIVEN a plant or system condition, PREDICT the operation of the Residual Heat Removal system

Reference: POP02-RH-0001 (Rev 48), note before step 6.3; LOT201.09, PowerPoint slides 11 & 12

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: A single wide range pressure does not feed all suction valves. The valves will not open even if the control switch is held on the open position.

B: INCORRECT: A single wide range pressure does not feed all suction valves.

C: INCORRECT: The valves will not open even if the control switch is held on the open position.

D: CORRECT: Each wide range pressure transmitter feeds 2 RHR suction valves. Transferring control to the ASP will effectively bypass the interlock and is allowed by procedure.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have knowledge of the interlocks for the RHR suction valves and apply that knowledge to determine the effect of holding the switch in the open position. The candidate must also have a knowledge of the procedural allowances found in the operating procedure.

Exam Bank No.: 1740 **Last used on an NRC exam:**

Unit 1 has experienced a Small Break Loss Of Coolant Accident. The Control Room Staff is performing the actions of 0POP05-EO-ES12, Post LOCA Cooldown and Depressurization, with the following plant condition:

- The TSC Staff has determined that hydrogen dissolution has occurred in the Pressurizer reference leg and that Pressurizer heaters should not be energized until Pressurizer level indication is 30-40%

Which one of the following correctly describes how indicated Pressurizer level may be affected AND the basis for the TSC Staff determination?

A. Indicated Pressurizer level may be greater than actual Pressurizer level. It is to ensure Pressurizer heaters are covered prior to being energized.

B. Indicated Pressurizer level may be less than actual Pressurizer level. It is to ensure Pressurizer heaters are covered prior to being energized

C. Indicated Pressurizer level may be greater than actual Pressurizer level. It is to ensure appropriate proportion of steam and water volumes to provide adequate cushioning during RCS pressure transients.

D. Indicated Pressurizer level may be less than actual Pressurizer level. It is to ensure appropriate proportion of steam and water volumes to provide adequate cushioning during RCS pressure transients.

Answer: A Indicated Pressurizer level may be greater than actual Pressurizer level. It is to ensure Pressurizer heaters are covered prior to being energized.

Exam Bank No.: 1740 **RO Outline Number:**

K/A Catalog Number: EPE E03EK2.1 **Tier:** 1 **Group/Category:** 2

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

Knowledge of the operational implications of the following concepts as they apply to the (LOCA Cooldown and Depressurization): Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

STP Lesson: LOT 504.12 **Objective Number:** 81517

Given a copy of a caution or note STATE/IDENTIFY the basis to include its purpose and the adverse impact of failure to comply with the caution or note.

Reference: 0POP05-EO-ES12 Post LOCA Cooldown and Depressurization (Rev 15), EOPT-03.09, WOG ERG ES-1.2, WOG ERG Background ES-1.2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: Hydrogen dissolution in the pressurizer reference leg will lower the amount of mass that exists in the reference leg and make indicated pressurizer level greater than actual level. The level requirement ensures that the pressurizer heaters are covered prior to energizing them.

B: INCORRECT: Hydrogen dissolution in the pressurizer reference leg will lower the amount of mass that exists in the reference leg and make indicated pressurizer level greater than actual level, not cause indicated level to be less than actual level.

C: INCORRECT: Hydrogen dissolution in the pressurizer reference leg will lower the amount of mass that exists in the reference leg and make indicated pressurizer level greater than actual level. Also, adequate Pressurizer pressure response does not become a concern unless level is much higher (reference TS 3.4.3)

D: INCORRECT: Hydrogen dissolution in the pressurizer reference leg will lower the amount of mass that exists in the reference leg and make indicated pressurizer level greater than actual level, not cause indicated level to be less than actual level. Also, adequate Pressurizer pressure response does not become a concern unless level is much higher (reference TS 3.4.3)

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have knowledge that a dissolution of hydrogen in the reference leg of the pressurizer will cause a loss of mass in the reference leg. The candidate must then apply the knowledge that this loss of mass in the reference leg will cause indicated pressurizer level to read greater than actual pressurizer level. The candidate must then understand that to ensure that the pressurizer heater are covered a higher pressurizer level is required.

Exam Bank No.: 1741 **Last used on an NRC exam:**

Which ONE of the following correctly describes the makeup water source(s) to the Component Cooling Water (CCW) Surge Tank?

A. Reactor Makeup Water via an automatic level control valve is the only makeup water source.

B. Demineralized Water via an automatic level control valve is the only makeup water source.

C. Reactor Makeup Water via an automatic level control valve is the normal makeup source and Demineralized Water via a manual valve is the backup source.

D. Demineralized Water via an automatic level control valve is the normal makeup source and Reactor Makeup Water via a manual valve is the backup source.

Answer: D Demineralized Water via an automatic level control valve is the normal makeup source and Reactor Makeup Water via a manual valve is the backup source.

Exam Bank No.: 1741 **RO Outline Number:**

K/A Catalog Number: 008 K1.05 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.0 **10CFR Reference:** 55.41(b)(8)

Knowledge of the physical connections and/or cause effect relationships between the CCWS and the following systems: Sources of makeup water.

STP Lesson: LOT 201.12 **Objective Number:** 5906

LIST all the systems that interface with the Component Cooling Water System and STATE the function of each interface.

Reference: LOT201.12 lesson handout page 10

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Demin water is the normal source with RMW the backup

B: INCORRECT: Demin water is the normal source with RMW the backup

C: INCORRECT: Demin water is the normal source with RMW the backup

D: CORRECT

Question Level: F **Question Difficulty** 2

Justification:

The candidate must have knowledge of system design which includes makeup water sources.

Exam Bank No.: 1742 **Last used on an NRC exam:**

Given the following plant conditions:

- A plant shutdown is in progress for refueling
- Permissive P-8 did not actuate as power was reduced

With Reactor power at 30%, which ONE of the following correctly describes the results of the P-8 malfunction?

A. The Reactor WILL trip if one Reactor Coolant Pump trips.

B. The Turbine will NOT trip if the Reactor trips.

C. The Reactor WILL trip if the Turbine trips.

D. The Reactor will NOT trip if Pressurizer level rises to 93%.

Answer: A The reactor WILL trip if one reactor coolant pump trips.

Exam Bank No.: 1742 **RO Outline Number:**

K/A Catalog Number: 012 K6.04 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(7)

Knowledge of the effect a loss or malfunction of the following will have on the RPS: Bypass-block circuits.

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 lesson handout #2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: A single loop loss of flow would normally be blocked by P-8 with power less than 40%

B: INCORRECT: A reactor trip always causes a turbine trip.

C: INCORRECT: Less than 50% power, a reactor trip is blocked when a turbine trip occurs by P-9 (which functioned correctly)

D: INCORRECT: The high pressurizer level trip is blocked by P-7 at less than 10% power.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the RPS blocks and be able to apply this knowledge to the given conditions to formulate a correct reponse.

Exam Bank No.: 1743 **Last used on an NRC exam:**

Given the following plant conditions:

- Reactor power is 65%
- SGFPs 11 and 12 are in service
- SGFP 13 is secured (0 rpm)

SGFP 12 trips and the secondary reactor operator notices:

- Startup SGFP failed to start
- All Steam Generator levels are lowering

Which of the following are immediate operator actions required FOR THESE CONDITIONS in accordance with POP04-FW-0002, Steam Generator Feed Pump Trip?

1. Attempt to manually start the Startup SGFP
2. Place SGFP 13 in service
3. Manually start the standby Feedwater Booster Pump
4. Check the SGFP Master Speed Controller properly responding in AUTO
5. Place all 4 Main Feedwater Regulating Valves in MANUAL

A. 1, 2, 3

B. 1, 3, 4

C. 2, 3, 4

D 3, 4, 5

Answer: B 1, 3, 4

Exam Bank No.: 1743 **RO Outline Number:**

K/A Catalog Number: 059 G2.4.49 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.6 **10CFR Reference:** 55.41(b)(7)

Main Feedwater System: Ability to perform without reference to procedures those action that require immediate operation of system components and controls.

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: POP04-FW-0002 (Rev 21), steps 1 and 2

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: SGFP 13 would only be placed in service if it was idling at 3300 rpm

B: CORRECT: Per POP04-FW-0002 step 1 RNO and step 2

C: INCORRECT: SGFP 13 would only be placed in service if it was idling at 3300 rpm

D: INCORRECT: Taking manual control of all MFRVs is not an immediate action for this condition and would likely make the transient more severe

Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze the given conditions to determine what has the highest priority and then must apply their knowledge of the off-normal procedure to formulate the correct response.

Exam Bank No.: 1744 **Last used on an NRC exam:**

Given the following condition:

- AFW Pump 11 is running with a flow rate of 540 gpm and a discharge pressure of 1550 psig

Which ONE of the following describes the system response as the AFW regulating valve for AFW Pump 11 is fully closed?

A. The Automatic Recirculation Valve opens to maintain pump flow and discharge pressure constant.

B. The Automatic Recirculation Valve opens to maintain pump flow constant. Pump discharge pressure will lower because back pressure from the AFW Storage Tank is less than that of the Steam Generator.

C. Pump flow will lower to about 100 gpm and stabilize as the Automatic Recirculation Valve opens. Discharge pressure will rise because the pump is operating closer to shutoff head conditions.

D. Pump flow will lower to about 100 gpm and stabilize as the Automatic Recirculation Valve opens. Discharge pressure will rise because the pump is operating closer to runout conditions.

Answer: C Pump flow will lower to about 100 gpm and stabilize as the Automatic Recirculation Valve opens. Discharge pressure will rise because the pump is operating closer to shutoff head conditions.

Exam Bank No.: 1744 **RO Outline Number:**

K/A Catalog Number: 061 K5.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.6 **10CFR Reference:** 55.41(b)(5)

Knowledge of the operational implications of the following concepts as they apply to the AFW: Pump head effects when control valve is shut.

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

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

Reference: LOT202.28 lesson, page 6

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: The auto recirc valve maintains a minimum flow, not normal flow. If flow remained constant then pressure would also remain constant.

B: INCORRECT: The auto recirc valve maintains a minimum flow, not normal flow. The pressure drop is across the valve, therefore discharge pressure measured upstream of the valve would not be dependant upon downstream pressure.

C: CORRECT: The auto recirc valve will maintain a minimum flow of approx 100 gpm. As the flow lowers, the pump approaches shutoff head conditions and the discharge pressure will subsequently rise.

D: INCORRECT: The auto recirc valve will maintain a minimum flow of approx 100 gpm, however runout describes a low discharge pressure condition, not a high discharge pressure condition.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have a knowledge of system/component design and operation. This knowledge along with a fundamental knowledge of pump operation must be combined to formulate the correct response.

Exam Bank No.: 1745 **Last used on an NRC exam:**

Given the following:

- Unit 2 is at 3% Reactor power performing a power ascension.
- Chemistry sampled the Refueling Water Storage Tank (RWST) and reports the boron concentration to be 2790 ppm.

Which one of the following correctly describes the results of the chemistry sample AND the appropriate action to take?

The reported RWST boron concentration is ...

A. BELOW Technical Specification required limits, the RWST should be processed through the Boron Thermal Regeneration System (BTRS) in the borate mode in accordance with 0POP02-CV-0002, Boron Thermal Regeneration System.

B. BELOW Technical Specification required limits, boric acid should be added to the RWST in accordance with 0POP02-CV-0001, Makeup to the Reactor Coolant System.

C. GREATER THAN Technical Specification required limits, the RWST should be processed through the Boron Thermal Regeneration System (BTRS) in the dilution mode in accordance with 0POP02-CV-0002, Boron Thermal Regeneration System.

D. GREATER THAN Technical Specification required limits, Reactor Makeup Water should be added to the RWST in accordance with 0POP02-CV-0001, Makeup to the Reactor Coolant System.

Answer: B BELOW Technical Specification required limits, boric acid should be added to the RWST in accordance with 0POP02-CV-0001, Makeup to the Reactor Coolant System.

Exam Bank No.: 1745 **RO Outline Number:**

K/A Catalog Number: 006 A2.10 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(5)

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: Low boron concentration in SIS.

STP Lesson: LOT 201.10 **Objective Number:** 4125

LIST all the systems that interface with the ECCS and state the function of each interface.

Reference: LOT201.10 (R15), ECCS, handout page 15; LOT201.10 Powerpoint slide 46, 0POP02-CV-0001 (R31), section 16

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Boron concentration is low. BTRS can add boron to the process stream, but is not designed to perform this function for the RWST and therefore is not addressed in the procedure.

B: CORRECT: Boron concentration is low and procedurally, the operators will add boric acid to the RWST to raise the boron concentration.

C: INCORRECT: Boron concentration is actually low. BTRS can remove boron from the process stream, but is not designed to perform this function and would only make the concentration lower.

D: INCORRECT: Boron concentration is actually low. Adding reactor makeup water will only make it lower.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must understand Tech Spec requirements for the RWST and then apply system knowledge to determine the method of correction based on the Tech Spec determination..

Exam Bank No.: 1746 **Last used on an NRC exam:**

If it is desired to cross-tie AFW Trains and reduce the number of running AFW Pumps 0POP01-ZA-0018, Emergency Operating Procedure Users Guide, allows this provided ...

A. Flow through a single AFW cross-tie valve must be less than 675 gpm.

B. Flow through a single AFW pump must be less than 675 gpm.

C. Flow through a single AFW cross-tie valve must be less than 640 gpm.

D. Flow through a single AFW pump must be less than 640 gpm.

Answer: B Flow through a single AFW pump must be less than 675 gpm.

Exam Bank No.: 1746 **RO Outline Number:**

K/A Catalog Number: 061 A1.05 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(5)

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

STP Lesson: LOT 504.04 **Objective Number:** T50404

Upon completion of this lesson, and without using reference material unless provided, the student will be able to explain the basic contents of the Emergency Operating Procedures

Reference: POP01-ZA-0018 (Rev 19), page 15

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: The limit is on flow through the pump, not flow through the valve.

B: CORRECT: To prevent pump runout, flow on a single AFW pump must not exceed 675 gpm

C: INCORRECT: The limit is on flow through the pump, not flow through the valve.

D: INCORRECT: The limit is 675 gpm. 640 gpm is the upper end of the automatic control range.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the flow limits on the AFW pumps to prevent runout.

Exam Bank No.: 1747 **Last used on an NRC exam:**

Which one of the following correctly describes entry conditions for 0POP05-EO-EC00, Loss of All AC Power?

A. 2 of 3 13.8 KV Standby Busses are de-energized.

B. All 13.8 KV Auxiliary AND Standby Busses are de-energized.

C. 2 of 3 Class 1E 4.16 KV ESF Busses are de-energized.

D. All Class 1E 4.16 KV ESF Busses are de-energized.

Answer: D All Class 1E 4.16 KV ESF Busses are de-energized.

Exam Bank No.: 1747 **RO Outline Number:**

K/A Catalog Number: 062 G2.4.2 **Tier:** 2 **Group/Category:** 1

RO Importance: 4.5 **10CFR Reference:** 55.41(b)(7)

AC Electrical Distribution: Knowledge of system setpoints, interlocks and automatic actions accociated with EOP entry conditions.

STP Lesson: LOT 504.22 **Objective Number:** 82074

STATE/IDENTIFY the two (2) entry conditions of POP05-EO-EC00 and any available control room indicators needed to make the determination.

Reference: POP05-EO-EC00 (Rev 19), page 1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Although the 13.8 KV busses can supply the class 1E busses, entry requirements for EC00 are based on the status of the Class 1E busses.

B: INCORRECT: Although the 13.8 KV busses can supply the class 1E busses, entry requirements for EC00 are based on the status of the Class 1E busses.

C: INCORRECT: All Class 1E busses must be de-energized to meet entry requirements of EC00.

D: CORRECT: All Class 1E 4.16 KV busses de-energized is one of the entry requirements for EC00.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the entry requirements for this EOP.

Exam Bank No.: 1748

Last used on an NRC exam:

Which of the following are 4.16 KV Bus 1D loads?

1. Start-up Feedwater Pump
2. Low Pressure Heater Drip Pumps
3. Open Loop Auxiliary Cooling Water Pumps
4. Reactor Containment Building Chillers
5. Instrument Air Compressors

A. 1, 2, 3

B. 2, 3, 4

C. 1, 4, 5

D. 2, 4, 5

Answer: B 2, 3, 4

Exam Bank No.: 1748 **RO Outline Number:**

K/A Catalog Number: 062 K2.01 **Tier:** 2 **Group/Category:** 1

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

Knowledge of bus power supplies to the following: Major system loads.

STP Lesson: LOT 201.31 **Objective Number:** 62355

STATE the function of the Electrical Distribution System and major components, interfaces and interlocks, physical arrangements, and the various system alignments.

Reference: LOT201.31, NON-CLASS 1E 13.8 TO 4160 VOLT AC DISTRIBUTION, PowerPoint slide #36

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: The essential chillers are supplied by class 1E 4KV power.

B: CORRECT: Typical non-class 4 KV loads

C: INCORRECT: The essential chillers are supplied by class 1E 4KV power and the feedwater booster pumps are a 13.8 KV load.

D: INCORRECT: The feedwater booster pumps are a 13.8 KV load.

Question Level: F **Question Difficulty** 3

Justification:

Candidate requires a basic knowledge of typical non-class 4.16 KV loads

Exam Bank No.: 1749 **Last used on an NRC exam:**

Given the following:

- ESF DG #12 is independently supplying 4.16 KV ESF Bus E1B.
- Bus voltage and frequency are stable.

Subsequently, a large load supplied by ESF Bus E1B is place in service.

Which one of the following correctly describes how the diesel will respond?

The diesel will slow down causing the speed governor to ...

A. adjust the fuel rack to allow the fuel injectors to inject more fuel.

B. adjust the fuel rack to raise the speed of the positive displacement fuel oil supply pump so it will pump more fuel.

C. throttle open the air intake butterfly valve allowing more combustion air to the engine.

D. raise the speed of the turbocharger and force additional combustion air into the engine.

Answer: A adjust the fuel rack to allow the fuel injectors to inject more fuel.

Exam Bank No.: 1749 **RO Outline Number:**

K/A Catalog Number: 064 K1.03 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(8)

Knowledge of the physical connections and/or cause-effect relationships between the ED/G system and the following systems: Diesel fuel oil supply system.

STP Lesson: LOT 201.39 **Objective Number:** 44273

DESCRIBE the flowpath of the Emergency Diesel Generator systems, sub systems, and interconnections with other systems

Reference: LOT201.39, ESF Diesel Generators, PowerPoint slide #63

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: Upon sensing a reduction in speed (frequency), the governor will move the fuel racks to allow more fuel to the engine.

B: INCORRECT: The fuel oil pump is a positive displacement pump. However fuel oil flow is controlled by the injectors.

C: INCORRECT: The function of the inlet butterfly is to shut off air, not control air during operation.

D: INCORRECT: Turbocharger speed is not "controlled", but is a function of load on the generator.

Question Level: F **Question Difficulty** 3

Justification:

The candidate requires a knowledge of the interaction between the diesel and the fuel oil sytem upon a change in generator load.

Exam Bank No.: 1750 **Last used on an NRC exam:**

The following plant conditions exist:

- Unit 1 is at 100% power
- A lockout occurs on 4.16 KV ESF Bus E1B.
- No operator actions are taken to change load on 125 VDC Bus E1B11.

If the E1B11 Battery discharge is allowed to continue, which of the following is correct?

E1B11 Battery voltage indication will........

A. continuously lower until the battery is fully discharged and then fall rapidly.

B. remain constant until the battery is fully discharged and then fall rapidly.

C. spike to an above-normal value during the initial minute of the discharge and then lower.

D. slowly rise during the last few hours of the discharge and then fall rapidly when the battery is fully discharged.

Answer: A continuously lower until the battery is fully discharged and then fall rapidly.

Exam Bank No.: 1750 **RO Outline Number:**

K/A Catalog Number: 063 A4.02 **Tier:** 2 **Group/Category:** 1

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

Ability to manually operate and/or monitor in the control room: Battery voltage indicator.

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

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: Battery voltage will decrease as the battery discharges until it is fully discharged.

B: INCORRECT: Battery voltage will not remain constant as the battery discharges. Battery voltage will decrease as the battery discharges.

C: INCORRECT: Battery voltage will not spike upwards. It will spike downwards during the initial stages of discharge.

D: INCORRECT: After the initial voltage spike, battery voltage will not rise. It will decrease as the battery discharges until it is fully discharged.

Question Level: F **Question Difficulty** 3

Justification:

The candidate requires the knowledge of battery discharge characteristics.

Exam Bank No.: 1751 **Last used on an NRC exam:**

With Unit 1 at 100% power, the following occurred:

- A 500 gpm RCS Loss of Coolant Accident (LOCA)
- A Loss of Offsite Power (LOOP).
- Essential Cooling Water (ECW) Pump 1B failed to start and will not start manually.
- The Control Room Staff is performing the actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection and has completed the actions of Addendum 5.
- No other manual operator actions have been taken.

Based on this information which ONE of the following is true?

A. Aux Feedwater Flow to Steam Generator 1B is being automatically controlled between 550 and 640 gpm.

B. Opening FV-7516, X CONN, for AFW Train 1B will feed Steam Generator 1B from a running Aux Feedwater Pump.

C. Aux Feedwater Pump #12 is running with approximately 100 gpm recirc flow.

D. Steam Generator 1B Aux Feedwater flow is 0 gpm. Aux Feedwater Pump #12 has no power available to it.

Answer: D Steam Generator 1B Aux Feedwater flow is 0 gpm. Aux Feedwater Pump #12 has no power available to it.

Exam Bank No.: 1751 **RO Outline Number:**

K/A Catalog Number: 076 K1.20 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(2)

Knowledge of the physical connections and/or cause-effect relationships between the SWS and the following systems: AFW.

STP Lesson: LOT 202.28 **Objective Number:** 43805

DESCRIBE the AFW system controls and instrumentation in the MCR.

Reference: 0POP05-EO-EO00, LOT202.28

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Aux Feedwater Pump #12 has lost power. No flow exists to Steam Generator 1B.

B: INCORRECT: Safety Injection Signal has not been reset yet so cross connect valves cannot be opened. I t also requires opening of at least 2 cross connect valves in order to feed one generator from an alternate train of AFW.

C: INCORRECT: Aux Feedwater Pump #12 has lost power. No flow exists through Aux Feedwater Pump #12.

D: CORRECT: Aux Feedwater Pump has lost power due to placing ESF Diesel Generator #12 in Pull To Stop due to the failure of ECW Pump 1B. Therefore flow from Aux Feedwater Pump 1B is 0 gpm and no Aux Feedwater flow is occurring to Steam Generator 1B.

Question Level: H **Question Difficulty** 3

Justification:

The candidate requires the knowledge that a LOOP concident with a loss of an ECW Pump will prevent re-energization of an ESF bus and loss of power to the AFW Pump power from that bus.

Exam Bank No.: 1752 **Last used on an NRC exam:**

The following plant conditions exist in Unit 1:

- Power Range Nuclear Instruments (NI) indicate 100% (all channels)
- Loop ΔT indicates 100% (all loops)
- Loop Tavg indicates 592°F (all loops)

If Tavg is lowered 1°F, which of the following is true? (Assume that Main Generator Load remains constant.)

A. Loop ΔT indication remains constant; Power Range NI indication will be slightly higher.

B. Loop ΔT indication remains constant; Power Range NI indication will be slightly lower.

C. Loop ΔT indication will be slightly lower; Power Range NI indication will remain constant.

D. Loop ΔT indication will be slightly lower; Power Range NI indication will be slightly lower.

Answer: B Loop delta T indication will remain constant; Power Range NI indication will be slightly lower.

Exam Bank No.: 1752 **RO Outline Number:**

K/A Catalog Number: 002 K5.10 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(5)

Knowledge of the operational implications of the following concepts as they apply to the RCS: Relationship between reactor power and RCS differential temperature.

STP Lesson: LOT 201.16 **Objective Number:** 96407

ANALYZE changes to plant conditions, instrument calibrations, and plant systems to determine effects on Excore Nuclear Instrumentation.

Reference: LOT201.16 Excore Nuclear Instrumentation System

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: If main generator load remains constant, core delta T required will remain constant even if Tavg goes down by one degree. However if Tavg goes down by one degree Tcold in the downcomer region of the Reactor Vessel goes down by one degree. This will cause fewer neutrons to reach the Power Range NI detectors causing indicated power to lower slightly, not go up slightly.

B: CORRECT: If main generator load remains constant, core delta T required will remain constant even if Tavg goes down by one degree. However if Tavg goes down by one degree Tcold in the downcomer region of the Reactor Vessel goes down by one degree. This will cause fewer neutrons to reach the Power Range NI detectors causing indicated power to lower slightly.

C: INCORRECT: If main generator load remains constant, core delta T required will remain constant even if Tavg goes down by one degree, not go down. However if Tavg goes down by one degree Tcold in the downcomer region of the Reactor Vessel goes down by one degree. This will cause fewer neutrons to reach the Power Range NI detectors causing indicated power to lower slightly, not remain constant.

D: INCORRECT: If main generator load remains constant, core delta T required will remain constant even if Tavg goes down by one degree, not go down. However if Tavg goes down by one degree Tcold in the downcomer region of the Reactor Vessel goes down by one degree. This will cause fewer neutrons to reach the Power Range NI detectors causing indicated power to lower slightly.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have the knowledge that core delta T is representative of power and will remain the same if power is constant. Also, that a decrease in Tcold in the downcomer area of the reactor vessel will reduce the number of neutrons reaching the Power Range Detectors causing an indicated lower power with the same core delta T.

Exam Bank No.: 1753 **Last used on an NRC exam:**

Given the following conditions:

- Unit 2 Reactor tripped from 100% power
- Tave is 573°F and slowly lowering

Which one of the following correctly identifies expected Steam Dump indications under these conditions?

STM DUMP UNBLOCK AVAILABLE white light is ...

A. LIT; UI-0555, DEMAND indicator is reading “0”.

B. LIT; UI-0555, DEMAND indicator is reading > “0”.

C. NOT lit; UI-0555, DEMAND indicator is reading “0”.

D. NOT lit; UI-0555, DEMAND indicator is reading >“0”.

Answer: B LIT; UI-0555, DEMAND indicator is reading > "0".

Exam Bank No.: 1753 **RO Outline Number:**

K/A Catalog Number: 041 A3.05 **Tier:** 2 **Group/Category:** 2

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

Ability to monitor automatic operation of the SDS, including: Main steam pressure,

STP Lesson: LOT 202.09 **Objective Number:** 93002

Given plant conditions, DETERMINE their effects on the Steam Dump System.

Reference: LOT202.09 (R12) Handout pages 9 and17

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Since Tave is grater than 567, there will be a demand signal generated.

B: CORRECT: A tubine trip armed the steam dumps causing the white light to be lit. Since Tave is abopve 567, there will be a demand signal.

C: INCORRECT: The dumps will be armed (white light on) due to the turbine trip. Since Tave is grater than 567, there will be a demand signal generated.

D: INCORRECT: The dumps will be armed (white light on) due to the turbine trip.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must determine plant/system conditions based on the given information and then apply that determination to Steam Dump operation to formulate the correct response.

Exam Bank No.: 1754 **Last used on an NRC exam:**

During normal plant operation, power is lost to RT-8032, GWPS Outlet Radiation Monitor.

Which one of the following describes the effects of the power loss and subsequent system operation?

The GWPS will ….

A. automatically shutdown. To re-establish GWPS operation, the waste gas flowpath must be re-aligned to bypass RT-8032. Grab samples will have to be taken periodically to monitor discharge radioactivity.

B. automatically shutdown. GWPS operation cannot be re-established until power is restored to RT-8032 unless the GWPS inlet and outlet valve controls are placed in the 'OPEN' position.

C. continue operating providing RT-8031, GWPS Inlet Radiation Monitor, remains in service because RT-8031 also provides for automatic isolation on high radioactivity.

D. continue operating, but grab samples will have to be taken periodically to monitor discharge radioactivity and the Oxygen Analyzer will be declared inoperable.

Answer: B automatically shutdown. GWPS operation cannot be re-established until power is restored to RT-8032 unless the GWPS inlet and outlet valve controls are placed in the 'OPEN' position.

Exam Bank No.: 1754 **RO Outline Number:**

K/A Catalog Number: 071 A2.05 **Tier:** 2 **Group/Category:** 2

RO Importance: 2.5 **10CFR Reference:** 55.41(b)(5)

Ability to (a) predict the impacts of the following malfunctions or operations on the Waste Gas Disposal System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Power failure to the ARM or PRM Systems.

STP Lesson: LOT 202.41 **Objective Number:** 11505

DESCRIBE the effect that a loss of the PERM system will have on radioactive effluent releases.

Reference: LOT202.41 student handout pages 26 and 27; 0POP02-WG-0001(R20)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: There is no bypass for this monitor.

B: CORRECT: The system will shutdown. Restoring the monitor or placing the valves in the OPEN position will allow operation of the system.

C: INCORRECT: The system will shutdown even with RT-8031 in service. RT-8031 has no automatic actions.

D: INCORRECT: The system will shutdown. The oxygen monitor will remain operable.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must understand the effects of a loss of power to a process radiation monitor with automatic actions and how it will affect operation of the GWPS.

Exam Bank No.: 1755 **Last used on an NRC exam:**

Unit 1 is operating at 100% power. The following conditions exist:

- ECW/CCW Train Mode Selector Switches are aligned as follows:
    - ‘A’ in RUN; ‘B’ in STBY; ‘C’ in OFF
- All ECW Pumps are running

An electrical failure causes a loss of power to 13.8 KV Standby Bus 1H.

ECW Pump 1C will be stripped and…..

A. must be started manually. Once started, it will automatically supply cooling water flow to #13 ESF D/G and Train ‘C’ CCW HX only.

B. then sequenced on, automatically supplying cooling water flow to #13 ESF D/G and Train ‘C’ CCW HX only.

C. must be started manually. Once started, it will automatically supply cooling water flow to #13 ESF D/G, Train ‘C’ CCW Hx, Train ‘C’ Essential Chiller and Train ‘C’ CCW Pump Supplementary Cooler.

D. then sequenced on, automatically supplying cooling water flow to #13 ESF D/G, Train ‘C’ CCW Hx, Train ‘C’ Essential Chiller, and Train ‘C’ CCW Pump Supplementary Cooler.

Answer: D sequenced on, automatically supplying cooling water flow to #13 ESF D/G, Train 'C' CCW Hx, Train 'C' Essential Chiller, and Train 'C' CCW Pump Supplementary Cooler.

Exam Bank No.: 1755 **RO Outline Number:**

K/A Catalog Number: 076 A3.02 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.7 **10CFR Reference:** 55.41(b)(7)

Ability to monitor automatic operation of the SWS, including: Emergency heat loads.

STP Lesson: LOT 201.13 **Objective Number:** 91201

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

Reference: LOT201.13 ECW System, LOT201.41 ESF Sequencers

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: ECW Pump 1C will be automatically sequenced on. Manual start of the pump is not required. When ECW Pump 1C is running it supplys cooling water flow to #13 ESF D/G, Train C CCW Hx, Train C Essential Chiller, and Train C CCW Supplemnetary Cooler.

B: INCORRECT: ECW Pump 1C is automatically sequenced on, however when it is running it will supply cooling water flow to #13 ESF D/G, Train C CCW Hx, Train C Essential Chiller, and Train C CCW Supplemnetary Cooler.

C: INCORRECT: ECW Pump 1C will be automatically sequenced on. Manual start of the pump is not required.

D: CORRECT: ECW Pump 1C will be automatically sequenced on. ESF D/G #13, Train C CCW HX, Train C Essential Chiller, and Train C CCW Pump Supplementary Cooler are all the loads that will be supplied. These loads are always alligned with manual valves to allow cooling water flow. Therefore when the ECW Pump is running cooling water flow is automatically supplied to the loads.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have an understanding of how a loss of the standby bus affects ECW operation along with a knowledge of the loads supplied by the ECW trains and that all loads are supplied when the ECW Pump in that respective train is running.

Exam Bank No.: 1756 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in Mode 6 with fuel offload in progress
- High radiation alarms are received on 68’ RCB Area Monitors RT-8099 and RT-8055
- 0POP04-FH-0001, Fuel Handling Accident, is entered

Which of the following are actions found in POP04-FH-0001 to protect the safety of site personnel under these conditions?

1. Ensure Control Room Envelope HVAC is operating in the Emergency Mode
2. Suspend core alterations in the Reactor Containment Building and the Fuel Handling Building
3. Manually initiate Safety Injection to actuate Containment Ventilation Isolation.
4. Establish Supplementary Purge using the Exhaust Fans only.
5. Initiate containment evacuation

A. 1, 2, 3

B. 2, 3, 4

C. 2, 4, 5

D. 1, 2, 5

Answer: D 1, 2, 5

Exam Bank No.: 1756 **RO Outline Number:**

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

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:** 92108

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

Reference: POP04-FH-0001 (Rev 13), steps 1, 3, and 4

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Although SI initiation will accomplish some of the required actions, other undesired consequences will also be seen which is why this is not required by procedure and should not be performed.

B: INCORRECT: Although SI initiation will accomplish some of the required actions, other undesired consequences will also be seen which is why this is not required by procedure and should not be performed. While establishing supplementary purge would correct the atmosphere in containment, it would result is a radiological release and should not be performed.

C: INCORRECT: While establishing supplementary purge would correct the atmosphere in containment, it would result is a radiological release and should not be performed.

D: CORRECT: Under the given conditions, these actions are performed per the off-normal procedure.

Question Level: F **Question Difficulty** 3

Justification:

The candidate requires a detailed knowledge of the off-normal procedure and must apply this knowledge to the given conditions to formulate the correct response.

Exam Bank No.: 1757 **Last used on an NRC exam:**

An Equipment Clearance Order (ECO) for removing power from the Jockey Fire Pump motor has been sent out to be hung.

The ECO places the handswitch in the OFF position and opens the supply breaker for the Jockey Fire Pump.

Based upon the actions taken when performing this lineup, what is the expected response of the Diesel Fire Pumps? (Assume that all Diesel Fire Pumps are available and aligned for automatic operation.)

A. No Diesel Fire Pumps will start.

B. All Diesel Fire Pumps will start 5 seconds after fire header pressure reaches 130 psig.

C. #11 Diesel Fire Pump will start 5 seconds after fire header pressure reaches 130 psig.

D. All Diesel Fire Pumps will start sequentially with a 5 second time delay between starts.

Answer: C #11 Diesel Fire Pump will start 5 seconds after fire header pressure reaches 130 psig.

Exam Bank No.: 1757 **RO Outline Number:**

K/A Catalog Number: 086 A1.05 **Tier:** 2 **Group/Category:** 2

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 Fire Protection System operating the controls including: FPS lineups.

STP Lesson: LOT 201.29 **Objective Number:** 53717

DESCRIBE the procedural requirements of POP02-FP-0001, Fire Protection System Operating Procedure, to include notes and precautions, and normal system alignment.

Reference: LOT201.29 Fire Protection System, 0POP02-FP-0001 Rev 25 Fire Protection System Operation

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: When the Jockey Fire Pump is taken to the OFF position and the supply breaker is opened, pressure in the fire header will decrease. This distractor assumes that the fire header will remain pressurized and not start the Diesel Fire Pumps.

B: INCORRECT: When the Jockey Fire Pump is taken to the OFF position and the supply breaker is opened, pressure in the fire header will decrease. The Diesel Fire Pumps each have different pressures and associated time delays at which they will start. #11 will start at 130 psig with a 5 second time delay. #12 will start at 120 psig with a 15 second time delay. #13 will start at 110 with a 25 second time delay. In this case only #11 Diesel Filer pump will start after pressure reaches 130 psig or below for 5 seconds. This distractor assumes that all the Diesel Fire Pumps have the same start setpoint and time delay.

C: CORRECT: When the Jockey Fire Pump is taken to the OFF position and the supply breaker is opened, pressure in the fire header will decrease. The Diesel Fire Pumps each have different pressures and associated time delays at which they will start. #11 will start at 130 psig with a 5 second time delay. #12 will start at 120 psig with a 15 second time delay. #13 will start at 110 with a 25 second time delay. In this case only #11 Diesel Filer pump will start after pressure reaches 130 psig or below for 5 seconds. This distractor is correct because when #11 Diesel Fire Pump starts it repressurizes the fire header and #12 and #13 Diesel Fire Pump start setpoints are never reached.

D: INCORRECT: When the Jockey Fire Pump is taken to the OFF position and the supply breaker is opened, pressure in the fire header will decrease. The Diesel Fire Pumps each have different pressures and associated time delays at which they will start. #11 will start at 130 psig with a 5 second time delay. #12 will start at 120 psig with a 15 second time delay. #13 will start at 110 with a 25 second time delay. In this case only #11 Diesel Filer pump will start after pressure reaches 130 psig or below for 5 seconds. This distractor assumes that all the Diesel Fire Pumps receive a start signal but will start them seqenntially at 5 second intervals.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the purpose of the fire water jockey pump and its effect on the system when it is secured. Must also have a knowledge of the pressure setpoints and time delays associated with these setpoints that automatically starts the diesel fire pumps.

Exam Bank No.: 1758 **Last used on an NRC exam:**

Given the following:

- Plant cooldown is in progress, currently in Mode 4 at 300 °F.
- Operators notice a rapid drop in Pressurizer level AND a rise in containment radiation levels.
- 0POP04-RC-0006, Shutdown LOCA, is entered.
- Due to continually lowering Pressurizer level, the procedure directs the operators to establish flow from the High Head Safety Injection pumps.

Based on these conditions, which ONE of the following correctly describes the availability of the High Head Safety Injection (HHSI) Pumps?

A. All 3 HHSI Pumps are immediately available for use.

B. 2 HHSI Pumps are immediately available for use. The breaker for the 3rd pump must be racked in prior to use.

C. 1 HHSI Pump is immediately available for use. The breaker for the standby pump must be racked in prior to use.

D. None of the HHSI Pumps are available for immediate use. The breaker for any pump to be started must be racked in first.

Answer: C 1 HHSI Pump is immediately available for use. The breaker for the standby pump must be racked in prior to use.

Exam Bank No.: 1758 **RO Outline Number:**

K/A Catalog Number: G2.4.9 **Tier:** 3 **Group/Category:**

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

Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies.

STP Lesson: LOT 506.01 **Objective Number:** 92161

DESCRIBE the general sequence of operation of components in the referenced procedure.

Reference: 0POP03-ZG-0007, Plant Cooldown (Rev 55), page 43

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: In Mode 4, all HHSI pumps cannot be operable.

B: INCORRECT: Although Tech Specs requires 2 operable HHSI pumps in Mode 4, it goes on to say that one of the operable pumps must have its breaker racked out.

C: CORRECT: Under the given plant conditions, 1 HHSI pump is required to be operable, a second available but with its breaker racked out.

D: INCORRECT: All HHSI pumps are required to be racked out in Mode 5, not Mode 4.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must use the given conditions and their knowledge of procedural requirements to determine the availability for use of the high head SI pumps under these circumstances.

Exam Bank No.: 1759 **Last used on an NRC exam:**

In accordance with 0POP01-ZA-0018, Emergency Operating Procedure User's Guide, which one of the following is correct regarding the initial performance of Emergency Operating Procedure immediate actions?

A. Each immediate action step is performed after being read by the Unit Supervisor.

B. All immediate action steps are performed after the RO obtains a working copy of the procedure.

C. Each immediate action step is performed after receiving a peer check from another operator.

D. All immediate action steps are performed immediately from memory.

Answer: D All immediate action steps are performed immediately from memory.

Exam Bank No.: 1759 **RO Outline Number:**

K/A Catalog Number: G2.4.13 **Tier:** 3 **Group/Category:**

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

Knowledge of crew roles and responsibilities during EOP usage.

STP Lesson: LOT 504.05 **Objective Number:** 80084

From memory STATE/IDENTIFY the immediate actions of POP05-EO-EO00 in their required sequence.

Reference: 0POP01-ZA-0018 (Rev 19), Emergency Operating Procedure User's Guide, Section 3.0 Responsibilities and Section 4.0 Rules of Usage

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Immediate action steps are performed immediately from memory. After they are performed from memory the Unit Supervisor will THEN read through each immediate action.

B: INCORRECT: Immediate action steps are performed immediately from memory. Working copies of EOPs are not used by the Reactor Operators.

C: INCORRECT: Immediate action steps are performed immediately from memory. Peer checks are not performed during the performance of immediate action steps.

D: CORRECT: Immediate action steps are performed immediately from memory.

Question Level: F **Question Difficulty** 3

Justification:

Candidate must have a knowledge of the requirement for performance of Emergency Operating Procedure Immediate Actions.

Exam Bank No.: 1760 **Last used on an NRC exam:**

In accordance with the Conduct of Operations Manual, which of the following are true with respect to the response to an UNEXPECTED alarm?

1. A member of the control room staff will announce the alarm title, with the communication being directed to the Unit Supervisor/Shift Manager.
2. A member of the control room staff will announce the alarm title, with the communication being directed to another Reactor Operator.
3. The Unit Supervisor/Shift Manager will provide a repeat back of the alarm received.
4. The other Reactor Operator will provide a repeat back of the alarm received.
5. All alarms are considered as UNEXPECTED the first time they are received unless they are the result of direct operator action.
6. The Unit Supervisor/Shift Manager shall scan the control panels for additional alarms when the alarm is acknowledged.

A. 1, 3, 5

B. 1, 3, 6

C. 2, 4, 5

D. 2, 4, 6

Answer: A 1, 3, 5

Exam Bank No.: 1760 **RO Outline Number:**

K/A Catalog Number: G2.4.31 **Tier:** 3 **Group/Category:**

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

Knowledge of annunciator alarms, indications, or response procedures.

STP Lesson: LOT 507.01 **Objective Number:** 92186

Given the title of an administrative procedure, DISCUSS the requirements associated with the referenced procedure.

Reference: Conduct of Operations, Chapter 2, pages 6 and 7

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: As required by the Conduct of Operations

B: INCORRECT: The board operators are responsible for scanning to find additional alarms

C: INCORRECT: The communication must be directed to, and acknowledged by a Unit/Shift Supervisor

D: INCORRECT: The communication must be directed to, and acknowledged by a Unit/Shift Supervisor. The board operators are responsible for scanning to find additional alarms

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the requirements found in the Conduct of Operations for alarm response in the control room.

Exam Bank No.: 1761 **Last used on an NRC exam:**

Unit 2 is in Mode 4, cooling down for a refueling outage.

Which one of the following would require entry into the action statement for Technical Specification 3.6.1.1, Containment Integrity?

A. The outer airlock door is opened for personnel entry.

B. LCV-0465, LTDN ISOL, is de-energized in the OPEN position for breaker maintenance.

C. MOV-0025, Charging OCIV, has a burned out motor and is currently de-energized in the CLOSED position.

D. The equipment Hatch was opened to allow moving outage equipment into the Containment Building.

Answer: D The equipment Hatch was opened to allow moving outage equipment into the containment building.

Exam Bank No.: 1761 **RO Outline Number:**

K/A Catalog Number: 103 K3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(7)

Knowledge of the effect that a loss or malfunction of the containment system will have on the following: Loss of containment integrity under shutdown conditions.

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, Tech Spec 3.6.1.1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: TS 3.6.1.3 allows opening the airlock doors one at a time for personnel entry.

B: INCORRECT: Although LCV-465 is an isolation valve by name, it is not a containment isolation valve.

C: INCORRECT: MOV-0025 is a containment isolation valve, but since it is de-energized closed, specification requirements are met.

D: CORRECT: By definition, cintainment integrity is not met with the equipment hatch open..

Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze each condition given with respect to the Tech Spec requirements to determine if the situation is in or out of compliance.

Exam Bank No.: 1813 **Last used on an NRC exam:**

The following plant conditions exist in Unit 1:

- Power is at 100%
- Most recent Reactor Coolant System Inventory indicates a small increase in the unidentified leakage rate.
- Preparations are being made to perform a Containment Building entry to determine the possible source of the leak.

Which ONE of the following is the greater potential radiological hazard to the personnel making entry and the method that could be used to reduce this hazard?

A. Iodine; Place Containment Carbon Units in service.

B. Iodine; Place Containment Normal Purge in service.

C. Radon; Place Containment Carbon Units in service.

D. Radon; Place Containment Normal Purge in service

Answer: A Iodine; Place Containment Carbon Units in service.

Exam Bank No.: 1813 **RO Outline Number:**

K/A Catalog Number: G2.3.14 **Tier:** 3 **Group/Category:** 3

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

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

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

A. Reactor Containment Fan Coolers
B. Containment Carbon Units
C. Control Rod Drive Mechanism Ventilation
D. Containment Cubicles Exhaust
E. Normal Containment Purge
F. Supplementary Containment Purge
G. Tendon Gallery Tunnel Ventilation
H. Reactor Cavity and Supports Ventilation
I. Elevator and Machinery Room Ventilation
J. RCB Chill Water
K. MSIV Cubicle Ventilatio

Reference: 0POP02-HC-0001, Containment HVAC, LOT202.33, Reactor Containment Building HVAC

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: Iodine is a greater radiological hazard than krypton due to its effect on the thyroid. The Containment Carbon Units can be placed in service to reduce these iodine levels.

B: INCORRECT: Iodine is a greater radiological hazard than krypton due to its effect on the thyroid. Containment Normal Purge cannot be placed in service while in Mode 1. It can only be used during Mode 5 or 6 or defueled conditon.

C: INCORRECT: Iodine is a greater radiological hazard than krypton due to its effect on the thyroid. The Containment Carbon Units can be placed in service to reduce these iodine levels.

D: INCORRECT: Iodine is a greater radiological hazard than krypton due to its effect on the thyroid. Containment Normal Purge cannot be placed in service while in Mode 1. It can only be used during Mode 5 or 6 or defueled conditon.

Question Level: H **Question Difficulty** 3

Justification:

Candidate must be able to distinguish between the greater radiological hazards associated iodine due to its effect on the thyroid and the radiological hazards posed by krypton. Must also be able to determine that the containment carbon units can be used to lower the iodine levels in the containment and that containment normal purge cannon be used in Mode 1, only Mode 5 or below.

Exam Bank No.: 1830 **Last used on an NRC exam:**

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 **RO Outline Number:**

K/A Catalog Number: G2.2.40 **Tier:** 3 **Group/Category:**

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(5)

Ability to apply Technical Specifications for a system.

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: New **Modified from**

Distractor Justification

A: INCORRECT: 2380 is the high pressure trip setpoint

B: INCORRECT: 2485 is the Pzr safety lift setpoint

C: CORRECT: Per Safety Limit 2.1.2, RCS pressure must not exceed 2735 psig.

D: INCORRECT: 3110 is the hydro criteria for the RCS.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the Safety Limits.

Exam Bank No.: 1831 **Last used on an NRC exam:**

With the Unit operating at full power, a single Shutdown Bank rod drops fully into the core.

Which one of the following describes the indications that will be seen by the operator and why?

The Rod Bottom Light for the affected rod will light due to a signal from the Rod Position Indication .....

A. reed switches; the ROD BOTTOM annunciator will NOT alarm because it only receives input from Control Bank rods.

B. reed switches; the ROD BOTTOM annunciator WILL alarm because it receives input from Shutdown Bank rods AND Control Bank rods.

C. coil stacks; the ROD BOTTOM annunciator will NOT alarm because it only receives input from Control Bank rods.

D. coil stacks; the ROD BOTTOM annunciator WILL alarm because it receives input from Shutdown Bank rods AND Control Bank rods.

Answer: D coil stacks; the ROD BOTTOM annunciator WILL alarm because it receives input from Shutdown Bank rods AND Control Bank rods.

Exam Bank No.: 1831 **RO Outline Number:**

K/A Catalog Number: 014 K4.03 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.2 **10CFR Reference:** 55.41(b)(6)

Knowledge of RPIS design feature(s) and/or interlocks which provide for the following: Rod bottom lights.

STP Lesson: LOT 201.19 **Objective Number:** 37816

STATE the location and DESCRIBE the principles of operation of the Bank Position Demand and Digital Rod Position Indication Systems.

Reference: LOT201.19, Rod Position Indicating System

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: DRPI does not utilize reed switches and any dropped rod will cause the annunciator to alarm.

B: INCORRECT: DRPI does not utilize reed switches

C: INCORRECT: Any dropped rod will cause the annunciator to alarm.

D: CORRECT: The DRPI system utilizes coil stacks and the rod bottom annunciator monitors all rods.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a basic knowledge of the design of the rod position indication system.

Exam Bank No.: 1833 **Last used on an NRC exam:**

Given the following:

- Unit 2 has a normal 13.8 KV electrical lineup.
- A compressor malfunction has resulted in one Starting Air Receiver on ESF DG #21 to completely depressurize.
- The second Starting Air Receiver is unaffected and at normal operating pressure.

Subsequently, a Unit 2 Standby Transformer lockout occurs.

Which one of the following correctly describes the effect of the depressurized air receiver on this event?

ESF DG #21 will...

A. NOT receive a start signal, but IS capable of starting if needed.

B. NOT receive a start signal and is NOT capable of starting.

C. receive a start signal and WILL start and run.

D. receive a start signal, but is NOT capable of starting.

Answer: C receive a start signal and WILL start and run.

Exam Bank No.: 1833 **RO Outline Number:**

K/A Catalog Number: 064 K6.07 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.7 **10CFR Reference:** 55.41(b)(7)

Knowledge of the effect of a loss or malfunction of the following will have on the ED/G System: Air Receivers

STP Lesson: LOT 201.39 **Objective Number:** 98476

Given a plant condition and/or various diesel modes of operation, PREDICT the response of the emergency diesels.

Reference: LOT201.39, ESF Diesel Generator, PowerPoint presentation slide #183

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: 4 KV Bus E1C will lose power resulting in a DG-13 start signal being generated. Second part is correct, diesel is capable of starting.

B: INCORRECT: 4 KV Bus E1C will lose power resulting in a DG-13 start signal being generated. Diesel can still start with 1 receiver depressurized.

C: CORRECT: 4 KV Bus E1C is normally powered from the U-1 Standby transformer, so when it is lost the diesel will receive a start signal. By design, each air receiver is capable of 5 consecutive starts without recharging.

D: INCORRECT: First part is correct, diesel will receive a start signal, however with 1 receiver depressurized, it can still start.

Question Level: H **Question Difficulty** 3

Justification:

Candidate must determine the effect of the loss on the transformer on the diesel and then determine the effect of the depressurized receiver on the start capability.

Exam Bank No.: 1836 **Last used on an NRC exam:**

Due to a malfunction, RT-8038, LWPS Discharge Monitor, erroneously indicates a HIGH alarm condition.

Which one of the following correctly describes the effect of the malfunction?

A. If the Condensate Polisher Cation Low Total Dissolved Solids (LTDS) tank is being discharged to the Neutralization basin, then the LTDS Transfer Pump will trip.

B. If the Condensate Polisher Cation Low Total Dissolved Solids (LTDS) tank is being discharged to the Neutralization basin, then FV-5804, TDS Waste Discharge to Neutralization Basin, will close.

C. If a Waste Monitor Tank (WMT) is being discharged to the Main Reservoir, then the Waste Monitor Tank Pump will trip.

D. If a Waste Monitor Tank (WMT) is being discharged to the Main Reservoir, then FV-4077, LWPS Discharge Valve will reposition to RECIRC.

Answer: D If a Waste Monitor Tank (WMT) is being discharged to the Main Reservoir, then
FV-4077, LWPS Discharge Valve will reposition to RECIRC

Exam Bank No.: 1836 **RO Outline Number:**

K/A Catalog Number: 073 K3.01 **Tier:** 2 **Group/Category:** 1

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

Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: Radioactive effluent releases.

STP Lesson: LOT 202.41 **Objective Number:** 11505

Describe the effect that a loss of the PERM system will have on radioactive effluent releases.

Reference: LOT202.41, Radiation Monitoring System, Handout #1, Page 26

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: RT-8042 monitors CP discharge to the neutralization basin. Second part is also incorrect in that the discharge valve would close instead of pump trip.

B: INCORRECT: RT-8042 monitors CP discharge to the neutralization basin. Second part would be correct concerning RT-8042.

C: INCORRECT: RT-8038 does moniotr WMT discharges, however upon high alarm the discharge valve moves to the recirc position instead of pump trip.

D: CORRECT: A high alarm on RT-8038 will cause FV-4077 to reposition to RECIRC.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a basic knowledge of the interface between RT-8038 and the system it monitors.

Exam Bank No.: 1842 **Last used on an NRC exam:**

Given the following Unit 1 conditions:

- A Large Break LOCA has occurred
- Operators have completed 0POP05-EO-EO00, Reactor Trip or Safety Injection, through step 4
- Containment pressure – 6.0 psig and stable
- Motor-driven AFW pumps – all running
- Turbine-driven AFW pump – not running
- Total AFW flow – 500 gpm
- Steam Generator NR levels range from 10 – 16%
- Addendum 5, Verification of SI Equipment Operation, is in progress at step 3, Verify AFW System Status.

As the Operator assigned to perform step 3 of Addendum 5, which one of the following correctly describes the MINIMUM required actions for step 3?

A. Align AFW Regulating valves to achieve greater than 576 gpm total AFW flow using ONLY the running Motor-driven pumps.

B. Align AFW Regulating valves to control ALL Steam Generator NR Levels between 14% and 50% with ONLY the running Motor-driven pumps.

C. Manually OPEN steam supply valves to start the Turbine-driven AFW pump then verify total AFW flow is greater than 576 gpm.

D. Manually OPEN steam supply valves to start the Turbine-driven AFW pump then control ALL Steam Generator NR Levels between 14% and 50%.

Answer: C Manually OPEN steam supply valves to start the Turbine-driven AFW pump then verify total AFW flow is greater than 576 gpm.

Exam Bank No.: 1842 **RO Outline Number:**

K/A Catalog Number: EPE 007 EA1.08 **Tier:** 1 **Group/Category:** 1

RO Importance: 4.4 **10CFR Reference:** 55.41(b)(10)

Ability to operate and monitor the following as they apply to a reactor trip: AFW System

STP Lesson: **Objective Number:** 80399

Reference: 0POP05-EO-EO00, Reactor Trip or Safety Injection, Rev. 20

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: If the TD AFWP is not in service, the step requires that it be manually started.

B: INCORRECT: due to adverse containment conditions, SG levels must be controlled to at least 34% NR.

C: CORRECT: Step 3 RNO has the Operator manually open steam supply valves to start the Turbine-driven AFW pump. Total AFW flow must be > 576 gpm because no SG levels are above the minimum of 34% (adverse containment).

D: INCORRECT: due to adverse containment conditions, SG levels must be controlled to at least 34% NR.

Question Level: H **Question Difficulty** 3

Justification:

requires candidate to determine appropriate actions based on given plant conditions including taking into account adverse containment conditions.

Exam Bank No.: 1843 **Last used on an NRC exam:**

Given the following:

- Unit 2 was operating at steady-state 100% power
- A Reactor trip occurred due to a Turbine trip
- RCS Tavg – 567 °F and stable
- Pressurizer pressure – 1920 psig and trending down slowly
- Pressurizer level – 36% and trending up slowly
- PRT level – 70% and stable
- PRT pressure – 4 psig and stable

Which one of the following describes current plant conditions; and required operator actions?

A. Parameters are trending to normal values after a Reactor trip; no actions necessary.

B. A leak has developed in the Pressurizer surge line; initiate Safety Injection to maintain RCS inventory.

C. A leak has developed on the spray line connection to the Pressurizer; raise Letdown flow to prevent overfilling the Pressurizer.

D. A leak has developed on the spray line connection to the Pressurizer; initiate Safety Injection to maintain RCS inventory.

Answer: D A leak has developed on the spray line connection to the Pressurizer; initiate Safety Injection to maintain RCS inventory.

Exam Bank No.: 1843 **RO Outline Number:**

K/A Catalog Number: APE 008 2.2.44 **Tier:** 1 **Group/Category:** 1

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

Pressurizer Vapor Space Accident, 2.2.44 – Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-EO00, Reactor Trip Or Safety Injection (Rev 20)
LOT201.14.HO.01, Pressurizer Pressure and level Control System Student Handout (Rev 12)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT – RCS pressure and temperature are not trending to normal values of 2235 psig and 25% level.

B: INCORRECT – Pressurizer level is trending up; not indicative of a RCS liquid leak

C: : INCORRECT – Correct leak location but wrong actions. Operator should initiate safety injection due to lowering pressurizer pressure.

D: CORRECT – A vapor space leak exists resulting in increasing pressurizer level and lowering pressure. Safety injection is required due to lowering pressurizer pressure.

Question Level: H **Question Difficulty** 3

Justification:

Requires the candidate to evaluate given plant conditions to determine plant status and appropriate action.

Exam Bank No.: 1844 **Last used on an NRC exam:**

Given the following conditions:

- Unit 1 is operating with a normal plant lineup at 100% power.
- An RCS leak/break occurs

Which one of the following correctly describes the MAXIMUM RCS inventory loss rate, and its basis, beyond which a Reactor Trip and Safety Injection would be required?

A. 240 gpm because the Charging Flow Control Valve, FCV-0205 can pass up to this much flow.

B. 200 gpm because the Charging Flow Control Valve, FCV-0205 can pass up to this much flow.

C. 200 gpm because the Reactor Makeup Control System can provide sufficient makeup to the VCT.

D. 240 gpm because the Reactor Makeup Control System can provide sufficient makeup to the VCT.

Answer: C 200 gpm because the Reactor Makeup Control System can provide sufficient makeup to the VCT.

Exam Bank No.: 1844 **RO Outline Number:**

K/A Catalog Number: EPE 009 EK3.03 **Tier:** 1 **Group/Category:** 1

RO Importance: 4.1 **10CFR Reference:** 55.41(b)(5)

Small Break LOCA, EK3.03 – Knowledge of the reasons for the following responses as they apply to the small break LOCA: Reactor trip and safety injection

STP Lesson: **Objective Number:**

Reference: LOT 201.06, 0POP04-RC-0003, RCS Leak

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: 240 gpm is the capacity of one CCP, not the charging flow control valve.

B: INCORRECT: 240 gpm is the capacity of one CCP and the charging flow control valve can pass > 400 gpm.

C: CORRECT: 200 gpm is the makeup capability t the VCT from the Rx MU Water Control System

D: INCORRECT: 240 gpm is the capacity of one CCP, not the MU capability to the VCT

Question Level: F **Question Difficulty** 3

Justification:

Requires knowledge of the makeup capability of CVCS

Exam Bank No.: 1846 **Last used on an NRC exam:**

Given the following:

- Unit 2 is at 100% power.
- Pressurizer Pressure Control is selected to position 457/456.
- Channel PT-0455 is bypassed for surveillance testing
- All Pressurizer Pressure controls are in AUTO.

As the Reactor Operator, you observe the following indications:

- PI-0456 – 1700 psig
- PI-0457 – 2200 psig
- PI-0458 – 2202 psig

Which one of the following is correct for the conditions described above?

A. "PRZR PRESS LO RX TRIP ALERT" alarm lit
Pressurizer Variable Heaters – Energized
Pressurizer Backup Heaters – De-energized

B. "PRZR PRESS LO RX TRIP ALERT" alarm lit
Pressurizer Variable Heaters – Energized
Pressurizer Backup Heaters – Energized

C. "PRZR PRESS LO PORV BLKD" alarm lit
Pressurizer Variable Heaters – Energized
Pressurizer Backup Heaters – De-energized

D. "PRZR PRESS LO PORV BLKD" alarm lit
Pressurizer Variable Heaters – Energized
Pressurizer Backup Heaters - Energized

Answer: B PRZR PRESS LO RX TRIP ALERT" alarm lit
Pressurizer Variable Heaters – Energized
Pressurizer Backup Heaters – Energized

Exam Bank No.: 1846 **RO Outline Number:**

K/A Catalog Number: APE 027 AA1.03 **Tier:** 1 **Group/Category:** 1

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

0027 Pressurizer Pressure Control System Malfunction, AA1.03 – Ability to operate and/or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: Pressure control when on a steam bubble

STP Lesson: **Objective Number:**

Reference: 0POP09-AN-04M8, Annunciator Lampbox 4M08 Response Instructions, Windows E-7 and F-8 (Rev 31)
LOT201.14.HO.02, Pressurizer Pressure Master Controller Student Handout (Rev 12)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: A: INCORRECT –“PRZR PRESS LO RX TRIP ALERT” would be lit due to channel 456 failing low however, Variable and Backup heaters would be energized when controlling channel (457) is below 2210 psig

B: B: CORRECT - “PRZR PRESS LO RX TRIP ALERT” would be lit due to channel 456 failing low. Variable and Backup heaters would be energized below 2210 psig on controlling channel (0457)

C: C: INCORRECT – “PRZR PRESS LO PORV BLKD” alarm would not be lit since its input is from channels 457 and 458 only and alarms at <2185 psig Also, Backup heaters would be energized below 2210 psig on controlling channel

D: D: INCORRECT - “PRZR PRESS LO PORV BLKD” alarm would not be lit since its input is from channels 457 and 458 only and alarms at <2185 psig

Question Level: H **Question Difficulty** 3

Justification:

Requires evaluating the given plant conditions to determine the status of the Pressurizer heaters and control room alarm.

Exam Bank No.: 1856 **Last used on an NRC exam:**

Given the following:

- Unit 2 has experienced a SG Tube Leak of 80 gallons per day (gpd)
- The crew is implementing 0POP04-RC-0004, Steam Generator Tube Leakage

Step 8.0 of 0POP04-RC-0004 requires the crew to check if BOTH of the following conditions exist:

- Leakage from any one SG is $\geq$ 75 gpd
- Rate of leakage increase $\geq$ 30 gpd/hr.

Which one of the below correctly describes the bases for monitoring these limits?

A. Leakage has exceeded Technical Specification limits for Primary-to-Secondary leakage and the unit should be rapidly shut down per 0POP03-ZG-0006, Plant Shutdown from 100% to Hot Standby.

B. Leakage has exceeded Technical Specification limits for Pressure Boundary leakage and the unit should be rapidly shut down per 0POP03-ZG-0006, Plant Shutdown from 100% to Hot Standby.

C. Leakage has increased to a size that indicates the unit should be shutdown in a planned manner by using 0POP03-ZG-0006, Plant Shutdown from 100% to Hot Standby.

D. These conditions are evidence the tube leak is increasing rapidly and the plant should be shutdown promptly by using 0POP04-TM-0005, Rapid Load Reduction.

Answer: D These conditions are evidence the tube leak is increasing rapidly and the plant should be shutdown promptly by using 0POP04-TM-0005, Rapid Load Reduction.

Exam Bank No.: 1856 **RO Outline Number:**

K/A Catalog Number: EPE 038 EK3.05 **Tier:** 1 **Group/Category:** 1

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

Steam Generator Tube Rupture, EK3.05 Knowledge of the reasons for the following responses as they apply to the SGTR: Normal operating precautions to preclude or minimize SGTR

STP Lesson: **Objective Number:**

Reference: Tech Spec 3.4.6.2, 0POP04-RC-0004, SG Tube Leakage

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Leakage is within TS limits

B: INCORRECT: Leakage is not considered pressure boundary leakage

C: INCORRECT: These conditions warrant a prompt plant shutdown per 0POP04-RC-0004, SG Tube Leakage

D: CORRECT: These conditions warrant a prompt plant shutdown per 0POP04-RC-0004, SG Tube Leakage

Question Level: F **Question Difficulty** 4

Justification:

must know the bases of the conditions for determining whether a prompt shutdown should be performed.

Exam Bank No.: 1857 **Last used on an NRC exam:**

The Caution prior to step 1 of 0POP05-EO-E020, Faulted Steam Generator Isolation, states:

"If any faulted SG or secondary break is NOT needed for RCS cooldown, THEN it should remain isolated during subsequent recovery actions to …"

A. minimize release to the containment or atmosphere.

B. prevent feeding a hot and dry steam generator.

C. prevent uncontrolled RCS cooldown.

D. prevent runout of the AFW pumps.

Answer: C prevent uncontrolled RCS cooldown.

Exam Bank No.: 1857 **RO Outline Number:**

K/A Catalog Number: APE 040 G2.4.20 **Tier:** 1 **Group/Category:** 1

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

Steam Line Rupture – Excessive Heat Transfer: 2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-E020, Faulted Steam Generator Isolation (Rev 9) LOT504.13.LP, Faulted Steam Generator Isolation (Rev 10)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Refer to Correct Answer

B: INCORRECT: Refer to Correct Answer

C: CORRECT: The caution states “If any faulted or secondary break is NOT needed for RCS cooldown, THEN it should remain isolated during subsequent recovery actions to prevent uncontrolled RCS cooldown.

D: INCORRECT: Refer to Correct Answer

Question Level: F **Question Difficulty** 3

Justification:

Must know what bases of caution is.

Exam Bank No.: 1858 **Last used on an NRC exam:**

Given the following:

- Unit 1 is operating at 100% power
- A Feedwater Line Break on SG 1C occurs inside the RCB.

Assuming no Operator action is taken, which one of the following describes the operational implications of this event?

A. RCS Tavg rises prior to the Reactor trip. The post-trip RCS cooldown will be the same as if a steamline break occurred on the same SG.

B. RCS Tavg rises prior to the Reactor trip. The post-trip RCS cooldown will be less than if a steamline break occurred on the same SG.

C. RCS Tavg lowers prior to the Reactor trip. The post-trip RCS cooldown will be the same as if a Steamline Break occurred.

D. RCS Tavg lowers prior to the Reactor trip. The post-trip RCS cooldown will be less than if a steamline break occurred on the same SG.

Answer: B RCS Tavg rises prior to the Reactor trip. The post-trip RCS cooldown will be less than if a steamline break occurred on the same SG.

Exam Bank No.: 1858 **RO Outline Number:**

K/A Catalog Number: APE 054 AK1.01 **Tier:** 1 **Group/Category:** 1

RO Importance: 4.1 **10CFR Reference:** 55.41(b)(8)

Loss of Main Feedwater, AK1.01 Knowledge of the operational implications of the following concepts as they apply to Loss of Main Feedwater (MFW): MFW line break depressurizes the S/G (similar to a steam line break)

STP Lesson: **Objective Number:**

Reference: LOT501.17, Decrease in Heat Removal by the Secondary System (Rev 3)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: RCS Tavg will rise due to lowering feed flow to the affected SG causing a decrease in heat transfer in that SG, but a FW line break creates less cooling on the RCS because the loss of FW from the SG and out the break does not involve the removal of latent heat as occurs in a steam break.

B: CORRECT: RCS Tavg will rise due to lowering feed flow to the affected SG causing a decrease in heat transfer in that SG. A FW line break creates less cooling on the RCS because the loss of FW from the SG and out the break does not involve the removal of latent heat as occurs in a steam break.

C: INCORRECT: RCS Tavg will rise due to lowering feed flow to the affected SG causing a decrease in heat transfer in that SG. Also, a FW line break creates less cooling on the RCS because the loss of FW from the SG and out the break does not involve the removal of latent heat as occurs in a steam break.

D: INCORRECT: RCS Tavg will rise due to lowering feed flow to the affected SG causing a decrease in heat transfer in that SG.

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine plant response on a FW line break at the specified location.

Exam Bank No.: 1859 **Last used on an NRC exam:**

Unit 2 is progressing through 0POP05-EO-EC00, Loss of All AC Power, and preparing to depressurize the intact SG's to 355 psig in accordance with step 16.

A Caution prior to step 16 indicates SG pressures should remain greater than 255 psig.

This Caution is based on:

A. Preventing injection of SI Accumulator nitrogen into the RCS which could impede natural circulation.

B. Ensuring a minimum of 35 °F subcooling is maintained for RCS pressure control.

C. Preventing void formation in the Reactor Vessel Head to prevent possible core uncovery.

D. Ensuring adequate RCP seal DP to allow for subsequent RCP operation.

Answer: A Preventing injection of SI Accumulator nitrogen into the RCS which could impede natural circulation.

Exam Bank No.: 1859 **RO Outline Number:**

K/A Catalog Number: EPE 055 EK1.02 **Tier:** 1 **Group/Category:** 1

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

Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-EC00, Loss of All AC Power (Rev 19)
ECA-0.0, Westinghouse Background Document (Rev 2, 4/30/2005)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: IAW with ECA-0.0, Westinghouse Background Document (Rev 2, 4/30/2005), the bases of the caution prior to step 16 is based on preventing injection of SI Accumulator nitrogen into the RCS which could impede natural circulation.

B: INCORRECT: Refer to Correct answer

C: INCORRECT: Refer to Correct answer

D: INCORRECT: Refer to Correct answer

Question Level: F **Question Difficulty** 3

Justification:

Must know bases of Caution

Exam Bank No.: 1860 **Last used on an NRC exam:**

Given the following:

- Unit 1 is operating at 100% power
- The following Annunciator and BYP INOP alarms are received in the Control Room
  - 120V AC CH I DIST PNL 001 TRBL
  - 125V DC SYSTEM E1A11 TRBL
  - BATT A11 OUTP BKR
  - INVRTR IV 001
  - 4KV E1A SPLY BKR TRIP
  - 4KV BUS O/C LCKOUT

Assuming no operator actions, which one of the below correctly describes the status of 120 VAC Panel DP-001?

DP-001 is ….

A. ENERGIZED because its Static Switch has automatically transferred to an alternate power source.

B. ENERGIZED because its Bypass Switch has automatically transferred to an alternate power source.

C. DE-ENERGIZED because its Rectifier/Inverter unit has lost power.

D. DE-ENERGIZED because its Voltage Regulating Transformer (VRT) has lost power.

Answer: C DE-ENERGIZED because its Rectifier/Inverter unit has lost power.

Exam Bank No.: 1860 **RO Outline Number:**

K/A Catalog Number: APE 057 AA2.06 **Tier:** 1 **Group/Category:** 1

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

0057 Loss of Vital AC Instrument Bus, AA2.06 – Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: AC instrument bus alarms for the inverter and alternate power source

STP Lesson: **Objective Number:**

Reference: 2POP-09-AN-03M2, Annunciator Lampbox 2-03M-2 Response Instructions (Rev 24), LOT 201.38, Class 1E 120 VAC
LOT201.38, Class 1E Vital 120 VAC (Rev 9)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: DP-001 distribution is not equipped with a Static Transfer sw., but DP 1201 is.

B: INCORRECT: DP-001 distribution is not equipped with a Bypass sw., but DP 1201 is. Additionally, the Bypass sw. is a manual device and will not automatically transfer power.

C: CORRECT: The combination of alarms provided would indicate there is a loss of power to DP-001.

D: INCORRECT: The VRT has lost power, but it normally doesn't supply DP-001. It requires a manual transfer to provide power.

Question Level: H **Question Difficulty** 3

Justification:

Must be able to evaluate the given alarms and determine the resulting condition of the power to Vital 120 VAC Bus DP-001.

Exam Bank No.: 1861 **Last used on an NRC exam:**

Given the following:

- Unit 2 was operating at steady-state 100% power.
- Train ‘B’ ECW and CCW is in service.
- A steam line break resulted in a Reactor trip and Safety Injection.
- Annunciator 02M4-D-1, CCW HX 2B OUTL TEMP HI alarms.

What is the appropriate operator action to address the alarm?

A. Verify proper ECW flow through the CCW/ECW Heat Exchanger.

B. Verify a CCW Level 2 isolation has occurred to isolate non-essential CCW heat loads.

C. Verify the ECW bypass valve on the CCW/ECW Heat Exchanger is throttled closed to maximize CCW/ECW heat transfer.

D. Verify the CCW bypass valve on the CCW/ECW Heat Exchanger is throttled open to reduce the heat load on the ECW system.

Answer: A Verify proper ECW flow through the CCW heat exchanger.

Exam Bank No.: 1861 **RO Outline Number:**

K/A Catalog Number: APE 062 AA1.07 **Tier:** 1 **Group/Category:** 1

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

Loss of Nuclear Service Water, AA1.07 - Ability to operate and/or monitor the following as they apply to the Loss of Nuclear Service Water: Flow rates to the components and systems that are serviced by the SWS; interactions among the components

STP Lesson: **Objective Number:**

Reference: 0POP09-AN-02M4, Annunciator Lampbox 2M04 Response Instructions (Rev 21)
LOT201.12, CCW System (Rev 13)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: This is the action required in the ARP.

B: INCORRECT: A Level 2 isolaton does not occur on a SI.

C: INCORRECT: There is no ECW bypass valve on the CCW/ECW Hx. There is a CCW bypass valve.

D: INCORRECT: Throttling open the CCW bypass valve will act to raise Hx outlet temperature more.

Question Level: H **Question Difficulty** 3

Justification:

Must determine appropriate action based on given information.

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

Given the following conditions:

- Unit 1 at 100% steady-state power.
- Instrument air pressure is 59 psig and trending down slowly.

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

A. Attempt to restore Instrument air pressure while monitoring primary and secondary systems for "valve drift"; Trip the Reactor upon loss of plant control.

B. Attempt to restore Instrument air pressure; Trip the Reactor if Instrument air pressure drops below 50 psig.

C. Attempt to restore Instrument air pressure; Trip the Reactor if Instrument air pressure drops below 55 psig.

D. Trip the Reactor and perform the actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Answer: D Trip the Reactor and perform the actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Exam Bank No.: 1863 **RO Outline Number:**

K/A Catalog Number: APE 065 A2.05 **Tier:** 1 **Group/Category:** 1

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

Ability to determine and interpret the following as they apply to the Loss of Instrument Air: When to commence plant shutdown if instrument air is decreasing

STP Lesson: **Objective Number:**

Reference: 0POP04-IA-0001, Loss of Instrument Air (Rev 13)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: The reactor is required to be tripped if air pressure goes below 60 psig.
B: INCORRECT: The reactor is required to be tripped if air pressure goes below 60 psig.
C: INCORRECT: The reactor is required to be tripped if air pressure goes below 60 psig.
D: CORRECT: Action required per Off Normal procedure.

Question Level: F **Question Difficulty** 3

Justification:

Must know the Off Normal procedure requirements

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

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 one of the following describes expected plant response and appropriate actions in accordance with 0POP05-EO-FRH1, Loss of Secondary Heat Sink, step 13?

A. RCS may not depressurize sufficiently to permit adequate SI flow to remove core decay heat. Open reactor vessel head vent valves.

B. RCS may not depressurize sufficiently to permit adequate SI flow to remove core decay heat. Close the open PORV to conserve RCS inventory and continue efforts to restore AFW flow.

C. RCS will not depressurize resulting in greater inventory loss. SI flow will be adequate for decay heat removal and no action is required.

D. RCS will not depressurize resulting in reduced SI flow due to higher equilibrium pressure. Close the open PORV to conserve RCS inventory and continue efforts to restore AFW flow.

Answer: A RCS may not depressurize sufficiently to permit adequate SI flow to remove core decay heat. Open reactor vessel head vent valves.

Exam Bank No.: 1864 **RO Outline Number:**

K/A Catalog Number: EPE E05 EK3.1 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.4 **10CFR Reference:** 55.41(b)(5)

Loss of Secondary Heat Sink, EK3.1 – Knowledge of the reasons for the following responses as they apply to the: Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-FRH1, Response to Loss of Secondary Heat Sink (Rev 17)
WOG ERG Background FRH.1 (Rev 2)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

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

B: INCORRECT: RCS may not depressurize sufficiently as stated, but action is incorrect.

C: INCORRECT: RCS will depressurize as stated, but action is incorrect.

D: INCORRECT: RCS will depressurize as stated, but action is incorrect.

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine plant response based on information provided and know purpose of opening Pzr PORV's and alternate action if PORV's do not open.

Exam Bank No.: 1865 **Last used on an NRC exam:**

Given the following:

- Unit 1 is operating at 75%
- Turbine EHC is in IMP IN
- Generator Voltage Regulator is ON (Auto)
- Severe Thunderstorms have caused a grid disturbance

Unit 1 Control Room receives the GEN MAX EXCT alarm on annunciator lampbox 7M01 window C-4.

Which one of the below describes the grid disturbance and required operator actions in response to the alarm condition?

A. Grid voltage has INCREASED causing the generator voltage regulator to raise excitation; LOWER excitation using the Voltage Adjuster control.

B. Grid voltage has DECREASED causing the generator voltage regulator to raise excitation; LOWER excitation using the Voltage Adjuster control.

C. Grid frequency has DECREASED causing the Main Turbine to shed MW load. LOWER Turbine MW load by using the Turbine EHC controls.

D. Grid frequency has INCREASED causing the Main Turbine to take more MW load. RAISE Turbine MW load by using the Turbine EHC controls.

Answer: B Grid voltage has DECREASED causing the generator voltage regulator to raise excitation; LOWER excitation using the Voltage Adjuster control.

Exam Bank No.: 1865 **RO Outline Number:**

K/A Catalog Number: APE 077 AK2.07 **Tier:** 1 **Group/Category:** 1

RO Importance: 3.6 **10CFR Reference:** 55.41(b)(5)

Generator Voltage and Electric Grid Disturbances, AK2.07 – Knowledge of the interrelations between Generator Voltage and Electric Grid Disturbances and the following: Turbine/Generator control

STP Lesson: **Objective Number:** 91615

Reference: 0POP09-AN-07M1, Annunciator Lampbox 7M01 Response Instructions (Rev 13)
LOT202.17.HO.02, Main Generator Turbine Generator and Exciter (Rev 9)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Increased grid voltage will cause the generator automatic voltage regulator (AVR) to attempt to lower grid voltage to setpoint. This will result in lower excitation field current that would not cause the alarm.

B: CORRECT: Decreased grid voltage would cause the AVR to attempt to raise voltage thereby increasing excitation field current and causing the alarm. Alarm response is to lower excitation field current using the Voltage Regulator..

C: INCORRECT: Grid frequency cannot affect turbine load because the EHC is in IMP IN mode. Additionally, changing load on the turbine is not an appropriate action for high excitation..

D: INCORRECT: Grid frequency cannot affect turbine load because the EHC is in IMP IN mode. Additionally, changing load on the turbine is not an appropriate action for high excitation..

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine which direction grid voltage changed in order for the voltage regulator to raise excitation automatically.

Exam Bank No.: 1866 **Last used on an NRC exam:**

Given the following:

- Unit 2 is operating at 100% power.
- The Control Room receives the ROD BOTTOM alarm on annunciator lampbox 5M03 window F-4
- The RO observes ONE dropped rod
- An I&C technician was dispatched to investigate and reports cause unknown.

Which one of the below describes the basis for reducing Reactor power LESS THAN 75% within 1 hour in accordance with 0POP04-RS-0001, Control Rod Malfunction?

A. Ensure sufficient margin to Axial Flux Difference limits while performing the dropped control rod recovery.

B. Ensure the Reactor Protection system will prevent exceeding core design criteria (during a transient) due to Nuclear Instrument shadowing induced by the dropped control rod

C. Ensure that Local Linear Heat Rate increases in other regions of the core (due to the dropped rod), will not cause core design criteria to be exceeded.

D. Ensure the increased power density due to Xenon decay within the affected fuel assembly (containing dropped rod) over the next few hours does not exceed core design criteria.

Answer: C Ensure that Local Linear Heat Rate increases in other regions of the core (due to the dropped rod), will not cause core design criteria to be exceeded.

Exam Bank No.: 1866 **RO Outline Number:**

K/A Catalog Number: APE 003 AK3.02 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(5)

Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Reactor Runback with a dropped control rod

STP Lesson: **Objective Number:** 91353

Reference: 0POP04-RS-0001, Control Rod Malfunction (Rev 20)
0POP09-AN-05M3, Annunciator Lampbox 5M03 Response Instruction (Rev 28)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: a single dropped rod will not have an appreciable effect on AFD. Mostly willl affect radial power (QPTR).

B: INCORRECT: While reducing power to 75% increases the margin to core design parameters, it does not provide protection in the event the plant is subjected to a transient. Step 11 of addendum 1 reduces RPS setpoints (if the dropped rod is expected to be greater than 2 hours for realignment) due to radial peaking.

C: CORRECT: per bases of step in Off Normal Procedure. Also, per bases of TS for rod alignment.

D: INCORRECT: Power density within the affected fuel assembly will decrease due to the effects of the control rod and Xe buildin within the time specified.

Question Level: F **Question Difficulty** 3

Justification:

Must know bases for reducing power to 75% to recover a dropped rod.

Exam Bank No.: 1867 **Last used on an NRC exam:**

Given the following:

- Unit 1 is operating at 100% power
- Pressurizer level is at 32%, trending down slowly
- 0POP04-RC-0004, Steam Generator Tube Leakage, procedure is being implemented.

The Crew should take which one of the following actions in accordance with 0POP04-RC-0004, Steam Generator Tube Leakage, to control Pressurizer level?

A. ISOLATE letdown flow; RAISE charging flow; START additional Charging Pump; TRIP the Reactor and INITIATE Safety Injection if Pressurizer level can NOT be maintained GREATER THAN 17%

B. ISOLATE letdown flow; RAISE charging flow; START additional Charging Pump; TRIP the Reactor and ALIGN CCP suction to the RWST if Pressurizer level can NOT be maintained GREATER THAN 17%

C. LOWER letdown flow; RAISE charging flow; START additional Charging Pump; TRIP the Reactor and INITIATE Safety Injection if Pressurizer level can NOT be maintained GREATER THAN 17%

D. LOWER letdown flow; RAISE charging flow; START additional Charging Pump; TRIP the Reactor and ALIGN CCP suction to the RWST if Pressurizer level can NOT be maintained GREATER THAN 17%

Answer: C LOWER letdown flow; RAISE charging flow; START additional Charging Pump; TRIP the Reactor and INITIATE Safety Injection if Pressurizer level can NOT be maintained GREATER THAN 17%

Exam Bank No.: 1867 **RO Outline Number:**

K/A Catalog Number: APE 037 AA1.11 **Tier:** 1 **Group/Category:** 2

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

Steam Generator Tube Leak, AA1.11 – Ability to operate and/or monitor the following as they apply to the Steam Generator Tube Leak: PZR level indicator

STP Lesson: **Objective Number:**

Reference: 0POP04-RC-0004, Steam Generator Tube Leakage (Rev 23)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Isolating Letdown flow is not a prescribed action as it would require isolation of charging or using charging and creating high thermal stress on the charging line connection.

B: INCORRECT: Isolating Letdown flow is not a prescribed action as it would require isolation of charging or using charging and creating high thermal stress on the charging line connection.

C: CORRECT: per the direction in 0POP04-RC-0004.

D: INCORRECT: Aligning CCP suction to the RWST is not a prescribed action for controlling Pressuizer level.

Question Level: F **Question Difficulty** 3

Justification:

Must know prescribed actions for controlling pressurizer level.

Exam Bank No.: 1868 **Last used on an NRC exam:**

Given the following:

- Unit 2 Control Room has been evacuated due to a fire.
- 0POP04-ZO-0001, Control Room Evacuation, is being implemented.
- RCP seal injection has been placed in service.
- Pressurizer level is 53% and trending up slowly.

Which one of the following describes the appropriate crew response in accordance with 0POP04-ZO-0001, Control Room Evacuation, to stabilize Pressurizer level?

A. Place Excess Letdown in service with flowpath to the Volume Control Tank.

B. Place Excess Letdown in service with flowpath to the Reactor Coolant Drain Tank.

C. Intermittently place Normal Letdown in service utilizing the 120 gpm orifice.

D. Place an additional Normal Letdown orifice in service to raise Letdown flow.

Answer: A Place Excess Letdown in service with flowpath to the Volume Control Tank.

Exam Bank No.: 1868 **RO Outline Number:**

K/A Catalog Number: APE 068 AA1.30 **Tier:** 1 **Group/Category:** 2

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

Ability to operate and/or monitor the following as they apply to the Control Room Evacuation: Operation of the letdown system

STP Lesson: **Objective Number:**

Reference: 0POP04-ZO-0001, Control Room Evacuation (Rev 30)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: action to be taken IAW 0POP04-ZO-0001, Control Room Evacuation
B: INCORRECT: Excess LD is not aligned to the RCDT
C: INCORRECT: Normal LD is isolated during Control Room Evac
D: INCORRECT: Normal LD is isolated during Control Room Evac

Question Level: F **Question Difficulty** 3

Justification:

Must know actions within procedure to stabilize Pressurizer level.

Exam Bank No.: 1869 **Last used on an NRC exam:**

Given the following:

- The crew has completed 0POP05-EO-EO20, Faulted SG Isolation, and have transitioned to 0POP05-EO-EO10, Loss of Reactor or Secondary Coolant
- All conditions are satisfied to terminate SI flow per step 15
- Current plant conditions are as follows:
  - o RCS subcooling - 45° F and stable
  - o Intact SG NR levels – 25% and stable
  - o Faulted SG level – 0% Wide Range
  - o RCS pressure – 1750 psig and stable
  - o Pressurizer level – 19%

Which one of the following describes the current status of the High Head Safety Injection system (HHSI)?

A. HHSI pumps are injecting into the RCS HOT legs and pressurizer level is trending up slowly

B. HHSI pumps are injecting into the RCS COLD legs and pressurizer level is trending up slowly

C. HHSI pumps are running at shutoff head with miniflow directed to the suction of each pump

D. HHSI pumps are running at shutoff head with miniflow directed to the RWST

Answer: D HHSI pumps are running at shutoff head with miniflow directed to the RWST

Exam Bank No.: 1869 **RO Outline Number:**

K/A Catalog Number: APE W/E02 EK1.1 **Tier:** 1 **Group/Category:** 2

RO Importance: 3.2 **10CFR Reference:** 55.41(b)(8)

Knowledge of the operational implications of the following concepts as they apply to the (SI Termination): Components, capacity, and function of emergency systems

STP Lesson: **Objective Number:**

Reference: 0POP-EO-EO00, Reactor Trip or Safety Injection (Rev 20)
LOT201.10.HO.01, Emergency Core Cooling Systems (Rev 15)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: RCS pressure is above the shutoff head of the HHSI pumps.

B: INCORRECT: RCS pressure is above the shutoff head of the HHSI pumps.

C: INCORRECT: RCS pressure is above the shutoff head of the HHSI pumps. Pump miniflow is directed back to the RWST, not the pump suciton.

D: CORRECT: RCS pressure is above the shutoff head of the HHSI pumps. Pump miniflow is directed back to the RWST.

Question Level: H **Question Difficulty** 3

Justification:

Must determine the status of the HHSI pumps based on a given set of plant conditions.

Exam Bank No.: 1870 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in Mode 4.
- RCS Pressure – 380 psig and stable.
- RCS Temperature - 330° F and stable.
- Train ‘A’ RHR is in service.
- Low Pressure Letdown from Train ‘A’ RHR is in service to the VCT.
- Low Pressure Letdown flow (LETDN HX OUTL FLOW FI-0132) – 130 gpm and stable.
- Normal Letdown is NOT in service.

Which ONE of the below describes the correct system response to a TRIP of the Train ‘A’ RHR Pump?

A. PRESS CONT PCV-0135 will automatically restore LETDN HX OUTL FLOW FI-0132 to 130 gpm.

B. RHR CVCS ISOL MOV-0066A will auto-close and secure Low Pressure Letdown.

C. Letdown OUTLET PRESS PI-0135 will lower, PRESS CONT PCV-0135 must be throttled CLOSED to raise pressure and restore Letdown flow to 130 gpm.

D. LETDN HX OUTL FLOW FI-0132 will lower, PRESS CONT PCV-0135 must be throttled OPEN to restore LETDN HX OUTL FLOW FI-0132 to 130 gpm.

Answer: D LETDN HX OUTL FLOW FI-0132 will lower, PRESS CONT PCV-0135 must be throttled OPEN to restore LETDN HX OUTL FLOW FI-0132 to 130 gpm.

Exam Bank No.: 1870 **RO Outline Number:**

K/A Catalog Number: 004 A4.05 **Tier:** 2 **Group/Category:** 1

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

Ability to manually operate and/or monitor in the control room: Letdown pressure and temperature control valves

STP Lesson: **Objective Number:**

Reference: 0POP02-CV-0004 Chemical and Volume Control System Subsystem (Rev 55)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: A loss of the RHR pump will cause letdown pressure to lower. PCV-0135 should be in manual, however, if it were in AUTO, it would close rather than open.

B: INCORRECT: There is no auto feature on MOV-0066A.

C: INCORRECT: PCV-0135 will have to be opened to raise LD flow.

D: CORRECT: A loss of the RHR pump will cause letdown pressure to lower. PCV-0135 will need to be opened to raise letdown flow.

Question Level: H **Question Difficulty** 3

Justification:

Must be able to determine how a loss of an RHR pump affects letdown flow in the given plant configuration and what action would be needed to restore letdown flow.

Exam Bank No.: 1871 **Last used on an NRC exam:**

Given the following:

- A Large Break Loss of Coolant Accident has occurred.
- The ECCS is operating in the Cold Leg Recirculation mode.

Based on standard heat transfer relationships (e.g. $\dot{Q} = UA\Delta T$), which one of the below describes the INITIAL effects of INCREASED CCW temperature to the RHR Heat Exchanger during these plant conditions?

ΔT across the RHR Heat Exchanger tubes will…

A. RISE resulting in LOWER heat transfer rate in the RHR Heat Exchanger.

B. RISE resulting in HIGHER heat transfer rate in the RHR Heat Exchanger.

C. LOWER resulting in LOWER heat transfer rate in the RHR Heat Exchanger.

D. LOWER resulting in HIGHER heat transfer rate in the RHR Heat Exchanger.

Answer: C LOWER resulting in LOWER heat transfer rate in the RHR Heat Exchanger.

Exam Bank No.: 1871 **RO Outline Number:**

K/A Catalog Number: 006 K5.11 **Tier:** 2 **Group/Category:** 1

RO Importance: 2.5 **10CFR Reference:** 55.41(b)(5)

Knowledge of the operational implications of the following concepts as they apply to ECCS: Basic heat transfer equation

STP Lesson: **Objective Number:**

Reference: LOT204.01, Integrated Plant Operations (Rev 1)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Delta-T across the Hx will go down resulting in less heat transfer.

B: INCORRECT: Delta-T across the Hx will go down resulting in less heat transfer.

C: CORRECT: Delta-T across the Hx will go down resulting in less heat transfer.

D: INCORRECT: Delta-T across the Hx will go down resulting in less heat transfer, not more heat transfer.

Question Level: H **Question Difficulty** 3

Justification:

Must be able to apply heat transfer relationships to the the given plant conditions.

Exam Bank No.: 1872 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in Mode 3
- A Pressurizer PORV has failed open and caused the PRT Rupture Disc to rupture.

Which of the below Containment parameters could be initially affected by this event AND indicates in the Control Room?

1. Air Temperature (RCFC Inlet)
2. ECCS Emergency Sump Level
3. RT-8011, RCS Atmosphere Rad Monitor
4. RT-8050, RCB High Range Area Monitor
5. Dewpoint

A. 1, 2, 3

B. 2, 3, 4

C. 1, 4, 5

D. 1, 3, 5

Answer: D 1, 3, 5

Exam Bank No.: 1872 **RO Outline Number:**

K/A Catalog Number: 007 K3.01 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(5)

Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment

STP Lesson: **Objective Number:**

Reference: LOT 201.01, LOT 202.41

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: The Emergency Sumps in containment do not have level indication.

B: INCORRECT: The Emergency Sumps in containment do not have level indication.

C: INCORRECT: The RCB High Range Rad monitor would not show any significant radiation level from normal RCS activity. This monitor has a 'live' zero of 1 Rem/hr.

D: CORRECT: Out of the 5 parameters listed, these 3 are the ones that may be affected. Refer to the distracter justifications for details.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must determine the effect of the stuck open PORV on the overall containment system, and then apply that determination to containment paramenters listed and determine if there will be an effect.

Exam Bank No.: 1873 **Last used on an NRC exam:**

Given the following:

- Unit 1 was operating at 100% power.
- A SGTR occurred in SG 1B.
- The crew is ready to depressurize the RCS in accordance with 0POP05-EO-EO30, SGTR.

Which one of the below correctly describe the basis for depressurizing the RCS?

A. Reduce RCP seal DP to minimize the possibility of a seal failure.

B. Ensure RCS pressure is within the limits of Tech Spec 3.4.9.1, Pressure/Temperature Limits.

C. Provide RCS inventory by re-filling the Pressurizer.

D. Prevent lifting the 1B SG Safety.

Answer: C Provide RCS inventory by re-filling the Pressurizer.

Exam Bank No.: 1873 **RO Outline Number:**

K/A Catalog Number: 010 2.4.18 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(5)

Knowledge of the specific bases for EOP's

STP Lesson: **Objective Number:**

Reference: ERG for 0POP05-EO-EO30, SGTR

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Refer to the Correct answer.
B: INCORRECT: Refer to the Correct answer.
C: CORRECT: per the Background document (ERG) for EO30.
D: INCORRECT: Refer to the Correct answer.

Question Level: F **Question Difficulty** 3

Justification:

Must know the bases of the depressurization step of EO30.

Exam Bank No.: 1875 **Last used on an NRC exam:**

Given the following:

- Unit 1 is preparing to conduct Main Steamline warm-up in accordance with 0POP03-ZG-0003, Secondary Plant Startup.
- Main Steamline Isolation has been reset.

Which one of the below sequences will allow throttling OPEN the Steam Generator 1B Main Steam Isolation Bypass (MSIB) Valve from the Control Room?

A.
1. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN B permissive grade solenoid to OPEN PERM.
2. Place handswitch SG 1B MSIB CONT FV-7422 in MODUL.
3. Throttle open SG 1B MSIB FV-7422 using Hagen controller.

B.
1. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN B permissive grade solenoid to OPEN PERM.
2. Throttle open SG 1B MSIB FV-7422 using Hagen controller.

C.
1. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN B permissive grade solenoid to OPEN PERM.
2. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN A permissive grade solenoid to OPEN PERM.
3. Throttle open SG 1B MSIB FV-7422 using Hagen controller.

D.
1. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN B permissive grade solenoid to OPEN PERM.
2. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN A permissive grade solenoid to OPEN PERM.
3. Place handswitch SG 1B MSIB CONT FV-7422 in MODUL.
4. Throttle open SG 1B MSIB FV-7422 using Hagen controller.

Answer: D Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN B permissive grade solenoid to OPEN PERM.
2. Place handswitch for Steam Generator 1B MAIN STEAM ISOL TRAIN A permissive grade solenoid to OPEN PERM.
3. Place handswitch SG 1B MSIB CONT FV-7422 in MODUL.
4. Throttle open SG 1B MSIB FV-7422 using Hagen controller.

Exam Bank No.: 1875 **RO Outline Number:**

K/A Catalog Number: 039 K4.08 **Tier:** 2 **Group/Category:** 1

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(7)

Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: Interlocks on MSIV and bypass valves

STP Lesson: **Objective Number:**

Reference: 0POP03-ZG-0003, Secondary Plant Startup (Rev 23)
LOT202.02, Main Steam (Rev 9)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT: Both train A and B control OPEN PERM switches must be momentarily placed in OPEN position to latch the open signal to the solenoids. This answer does not open the Train A solenoid.

B: INCORRECT: Same reason as A.

C: INCORRECT: MSIB control switch must be momentarily placed in MODUL to allow operation of the Hagen controller. This answer does not place the control handswitch to MODUL.

D: CORRECT: Train A and B permissive solenoids are energized and controller FV-7422 is placed in modulate. The Hagen controller is then allowed to modulate the MSIB.

Question Level: F **Question Difficulty** 3

Justification:

Must know sequence of switch operation based to enable the MSIB Hagan controller.

Exam Bank No.: 1876 **Last used on an NRC exam:**

Given the following:

- Unit 1 is at 90% reactor power
- Control rod C9 in Shutdown Bank B has dropped and is being recovered in accordance with 0POP04-RS-0001, Control Rod Malfunction, Addendum 1, Recovery of a Dropped Rod
- Lift coil disconnect switches for all rods in Shutdown Bank B, except rod C9, have been placed in the DISCONNECTED position

Which one of the following describes (1) where the lift coil disconnect switches are located and (2) the expected Control Room alarm as a result of disconnecting Shutdown Bank B control rods?

A. Lift coil disconnect switches are located in their respective rod control power cabinets; A ROD CONT URGENT ALARM will actuate when lift coil disconnect switches are placed in the DISCONNECT position.

B. Lift coil disconnect switches are located in their respective rod control power cabinets; A ROD CONT URGENT ALARM will actuate when rod withdrawal begins for control rod C9.

C. Lift coil disconnect switches are located in the Control Room; A ROD CONT URGENT ALARM will actuate when lift coil disconnect switches are placed in the DISCONNECT position.

D. Lift coil disconnect switches are located in the Control Room; A ROD CONT URGENT ALARM will actuate when rod withdrawal begins for control rod C9.

Answer: D Lift coil disconnect switches are located in the Control Room; A ROD CONT URGENT ALARM will actuate when rod withdrawal begins for control rod C9.

Exam Bank No.: 1876 **RO Outline Number:**

K/A Catalog Number: 001 A4.06 **Tier:** 2 **Group/Category:** 2

RO Importance: 2.9 **10CFR Reference:** 55.41(b)(6)

Ability to manually operate and/or monitor in the control room: Control rod drive disconnect/connect

STP Lesson: **Objective Number:**

Reference: 0POP04-RS-0001, Control Rod Malfunction (Rev 20)
LOT201.18 Rod Control (Rev 11)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: INCORRECT:All lift coil disconnect switches are located in the control room and not rod control power cabinets. Rod control urgent alarm will illuminate when shutdown bank B rod movement is demanded, not when lift coil disconnect switch is placed in disconnect.

B: INCORRECT: All lift coil disconnect switches are located in the control room.

C: INCORRECT: Rod control urgent alarm will not illuminate until rod movement is demanded.

D: CORRECT: Lift coil disconnect switches are located in the control room. Rod control urgent alarm will illuminate when shutdown bank B rod movement is demanded.

Question Level: F **Question Difficulty** 3

Justification:

must know where the disconnect switches are located and when the Rod Control Urgent Alarm will annunciate.

Exam Bank No.: 1877 **Last used on an NRC exam:**

Which one of the below correctly indicates the power supply and distribution for Power Range Channel N-42?

	Power Supply Distribution Panel (DP)	Power to Channel Ion Chamber Detector	Power to Channel Bistables
A	1202	Instrument Power	Control Power
B	002	Control Power	Instrument Power
C	002	Instrument Power	Control Power
D	1202	Control Power	Instrument Power

Answer: A 1202□Instrument Power□Control Power

Exam Bank No.: 1877 **RO Outline Number:**

K/A Catalog Number: 015 K2.01 **Tier:** 2 **Group/Category:** 2

RO Importance: 3.3 **10CFR Reference:** 55.41(b)(7)

Knowledge of bus power supplies to the following: NIS channels, components, and interconnections

STP Lesson: **Objective Number:**

Reference: LOT201.16.HO.01, Excore Nuclear Instrumentation (Rev 13)
0POP04-NI-0001, Nuclear Instrument Malfunction (Rev 18)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: Power supplies are as stated.

B: INCORRECT: NI42 recieves power from DP 1202, Control power supplies the bistables, instrument power supplies the detector.

C: INCORRECT: NI42 recieves power from DP 1202,

D: INCORRECT: Control power supplies the bistables, instrument power supplies the detector.

Question Level: F **Question Difficulty** 3

Justification:

Must know the powe distribution for the NI channels including what instrument and control power go to.

Exam Bank No.: 1879 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in Mode 5
- Preparations to enter mode 6 are in progress
- RCS boron concentration is 2830 ppm

Which one of the below describes (1) RCS Temperature requirements for entering Mode 6 and (2) when Mode 6 is declared?

A. RCS Tave $\leq$ 140 °F; mode 6 is declared when ANY Reactor Vessel head closure bolt is less than fully tensioned

B. RCS Tave $\leq$ 200 °F; mode 6 is declared when ANY Reactor Vessel head closure bolt is less than fully tensioned

C. RCS Tave $\leq$ 140 °F; mode 6 is declared when ALL Reactor Vessel head closure bolts are detensioned

D. RCS Tave $\leq$ 200 °F; mode 6 is declared when ALL Reactor Vessel head closure bolts are detensioned

Answer: A RCS Tave ≤ 140 °F; mode 6 is declared when ANY reactor vessel head closure bolt is less than fully tensioned

Exam Bank No.: 1879 **RO Outline Number:**

K/A Catalog Number: G2.2.35 **Tier:** 3 **Group/Category:** 2

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

Ability to determine Technical Specification Mode of Operation

STP Lesson: **Objective Number:**

Reference: STP Technical Specifications
0POP03-ZG-0010, Refueling Operations (Rev 50)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified from**

Distractor Justification

A: CORRECT: per TS definition of plant Modes

B: INCORRECT: Mode 6 is entered when RCS temperature is at or below 140 deg F, not 200 deg F.

C: INCORRECT: Mode 6 is declared when ANY RV head closure bolt is less than fully tensioned.

D: INCORRECT: Mode 6 is entered when RCS temperature is at or below 140 deg F, not 200 deg F.

Question Level: F **Question Difficulty** 3

Justification:

Must know the TS definitions of plant Modes.

Exam Bank No.: 587 **Last used on an NRC exam:** 1995

As the Core Loading Supervisor, which one of the following conditions would require you to immediately suspend further core alterations in accordance with 0POP08-FH-0009, Core Refueling?

A. As a fuel assembly is loaded, the count rate on SR channel N-31 increases from 8 cps to 16 cps, however the count rate on N-32 remains stable at 8 cps

B. The operating RHR loop inlet temperature has increased from 122°F to 130°F since the last recorded reading

C. Boron concentration in the operating RHR loop has decreased from 2803 ppm to 2797 ppm since the last sample

D. Moving the refueling bridge to a different location causes radiation levels on the bridge to increase from 7 mr/hr to 53 mr/hr

Answer: C Refueling canal boron concentration has decreased from 2803 ppm to 2797 ppm since the last sample.

Exam Bank No.: 587 **SRO Outline Number:**

K/A Catalog Number: G2.1.36 **Tier:** 3 **Group/Category:** 1

SRO Importance: 4.1 **10CFR Reference or SRO Objective:** 55.43(b)(6)

Knowledge of procedures and limitations involved in core alterations

STP Lesson: **Objective Number:** 60040

Discuss the requirements of the Core Refueling, 0POP08-FH-0009, to include: A. Prerequisites; B. Notes and Precautions; C. Major Procedural Steps; D. Checklists.

Reference: 0POP08-FH-0009, Rev 14, Core Refueling

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified From**

Distractor Justification

A: INCORRECT - Core alterations must only be suspended if both SR channels increase by a factor of 2, or a single channel increases by a factor of 5.

B: INCORRECT - Core alterations also be suspended if RHR temperature increases by 10 degrees-F since the previous reading.

C: CORRECT - Minimum required boron concentration in the operating RHR loop is 2800 ppm.

D: INCORRECT - The higher dose rate is due to bridge proximity to the vessel head and time must be minimized in this condition, but core alterations are not to be suspended.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the procedural requirements for continued fuel movement.

Exam Bank No.: 1147 **Last used on an NRC exam:** 2003

Given the following:

- Unit 1 is at 100% power with 79 days remaining until a refueling outage.
- Safety Injection Accumulator 1C is experiencing increased RCS in-leakage and requires draining twice per shift.
- A work package has been developed to shut the Accumulator 1C discharge isolation valve, MOV-0039C.
    - MOV-0039C valve control circuitry will be modified to receive an open signal on low RCS pressure in addition to a Safety Injection actuation signal
- MOV-0039C will be repaired and its control circuitry returned to original design configuration during the refueling outage.

What are the requirements for this modification to be implemented?

Since this modification ....

A. will be in service for less than 90 days, process this work package through the temporary modification process without performing a 10CFR50.59 evaluation.

B. may affect the Safety Injection function, process this work package through the 10CFR50.59 evaluation process.

C. establishes plant conditions that are in compliance with Technical Specifications, only a 10CFR50.59 screening is required.

D. meets the same functional requirements as the original design, the work package can be approved for implementation by performing only a 10CFR50.59 screening.

Answer: B Since this modification may affect the Safety Injection function, process this work package through the 10CFR50.59 evaluation process.

Exam Bank No.: 1147 **SRO Outline Number:**

K/A Catalog Number: G2.2.5 **Tier:** 3 **Group/Category:** 2

SRO Importance: 3.2 **10CFR Reference or SRO Objective:** 55.43(b)(3)

Knowledge of the process for making design or operating changes to the facility

STP Lesson: **Objective Number:**

Reference: 0PGP03-ZO-0003, Temporary Modifications (Rev 24)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: Bank **Modified From**

Distractor Justification

A: INCORRECT - 90 day limit only applies to temporary changes made to support maintenance activities.

B: CORRECT - Since this a compensatory measure to address a degraded condition, 50.59 should be applied.

C: INCORRECT - Established conditions will not satisfy Tech Specs, a 50.59 evaluation will be required.

D: INCORRECT - Same functional requirements are not met as the valve will remain closed (vice de-energized open) as original design dictates and a 50.59 evaluation is required.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must apply their knowledge of the 50.59 process and the temp mod process to the given scenario.

Exam Bank No.: 1762 **Last used on an NRC exam:**

The following plant conditions exist in Unit 1:

- Reactor power is at 48%
- Annunciator 7M3-E7, MAIN COND VACUUM LO, is in alarm.
- Condenser vacuum is 21 inches Hg and slowly LOWERING.

Based on these plant conditions, which ONE of the following describes the correct actions for the Unit Supervisor to take?

A. Commence a plant shutdown in accordance with 0POP03-ZG-0006, Plant Shutdown From 100% to Hot Standby.

B. Perform a fast load reduction in accordance with 0POP04-TM-0005, Fast Load Reduction.

C. Ensure the reactor and main turbine are tripped and perform the actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection

D. Trip the main turbine and perform the actions of 0POP04-TM-0003, Turbine Trip Below P-9.

Answer: C Ensure the reactor and main turbine are tripped and perform the actions of 0POP05-EO-EO00, Reactor Trip or Safety Injection.

Exam Bank No.: 1762 **SRO Outline Number:**

K/A Catalog Number: APE 051 AA2.02 **Tier:** 1 **Group/Category:** 2

SRO Importance: 4.1 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip.

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-CR-0001, Loss of Condenser Vacuum, Rev 12

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Due to lowering condenser vacuum, 0POP04-CR-0001, Loss of Condenser Vacuum would have been entered. 0POP03-ZG-0006, Plant Shutdown From 100% to Hot Standby, is not referenced or directed to be used to reduce load. At 21 inches vacuum the main turbine would have tripped and would have required a manual reactor trip and entry into 0POP05-EO-EO00, Reactor Trip or Safety Injection.

B: INCORRECT: 0POP04-CR-0001, Loss of Condenser Vacuum does reference use of 0POP04-TM-0005, Fast Load Reduction, however at 21 inches of vacuum the main turbine would have tripped. Therefore use of the Fast Load Reduction off-normal is not applicable.

C: CORRECT: At 21 inches vacuum the main turbine would trip. Actions contained in 0POP04-CR-0001, Loss of Condenser Vacuum require the operator to ensure that the reactor is tripped (with this set of conditions a manual reactor trip would be required), ensure that the main turbine is tripped (tripped when condenser vacuum reached 21 inches), and go to 0POP05-EO-EO00, Reactor Trip or Safety Injection.

D: INCORRECT: With this set of plant conditions the main turbine would automatically trip At 21 inches vacuum and not cause a reactor trip. However 0POP04-CR-0001, Loss of Condenser Vacuum requires that the reactor be tripped under all circumstances when condenser vacuum reaches 21 inches. Therefore entry into 0POP04-TM-003, Turbine Trip Below P-9 is not appropriate.

Question Level: H **Question Difficulty** 4

Justification:

Requires the candidate to know that the main turbine will automatically trip at 21 inches vacuum but will not cause an automatic reactor trip because the initial plant conditions places the plant at less than 50% (below P-9 setpoint). However it requires the candidate to know that the requirements contained in 0POP04-CR-0001, Loss of Condenser Vacuum require that the reactor be tripped when condenser vacuum reaches 21 inches under all circumstances.

Exam Bank No.: 1763 **Last used on an NRC exam:**

A Reactor trip has occurred due to a Loss of Offsite Power (LOOP) and the crew has reached step 16 of POP05-EO-ES02, Natural Circulation Cooldown, to initiate RCS depressurization.

Plant conditions are as follows:

- One CRDM fan is running
- CET temperature in 500° F
- RCS pressure is 1200 psig
- Upper head temperature is 520° F
- Auxiliary spray valve is closed

Based on these conditions, what is the NEXT action the Unit Supervisor should direct the crew to take?

A. Raise RCS pressure to establish greater than 85° F subcooling based on core exit T/Cs.

B. Ensure Normal Spray Valves are closed and then open the Auxiliary Spray Valve.

C. Raise RCS pressure to establish greater than 100° F subcooling based on core exit T/Cs.

D. Open the Reactor Vessel Head Vent Isolation Valves and one Reactor Vessel Head Vent Throttle Valve.

Answer: C Raise RCS pressure to establish greater than 100° F subcooling based on core exit T/Cs.

Exam Bank No.: 1763 **SRO Outline Number:**

K/A Catalog Number: EPE E09 G2.1.25 **Tier:** 1 **Group/Category:** 2

SRO Importance: 4.2 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Natural Circulation Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

STP Lesson: LOT 504.25 **Objective Number:** 92234

Given a copy of a step from 0POP05-EO-ES02, STATE/IDENTIFY how the action is performed and the basis for the action to include the action itself, its purpose and the result.

Reference: POP05-EO-ES02, Rev 12, step 16, addendums 1, 2, and 3

Attached Reference ☑ **Attachment:** POP05-EO-ES02, step 16, addendums 1, 2, and 3

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: This is the action for step 16.b, which is innapropriate since only 1 CRDM fan is running.

B: INCORRECT: This would be the correct choice if the candidate innapropriately determined subcooling criteria were met.

C: CORRECT: Based on the indications given, subcooling per Addendum 2 is not met and must be established per RNO 16.a.1

D: INCORRECT: This would be the correct choice if the Addendum 2 criteria is met and upper head subcooling is less than 10 degrees.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must assess the given conditions using the procedure given and determine the correct action to be taken.

Exam Bank No.: 1764 **Last used on an NRC exam:**

The following plant conditions exist in Unit 1:

- The Control Room Staff are in the process of draining the RCS to mid-loop conditions in accordance with 0POP03-ZG-0009, Mid-Loop Operation.
- Pressurizer level is 5%.
- RHR Train A is in service.
- RHR Trains B and C are in standby.
- RHR Train A Inlet Temperature is being maintained less than 140°F.
- Annunciator 1M2-F4, RHR PUMP 1A DISCH FLOW LO alarms.
- All automatic actions have occurred, as designed.

Based on these plant conditions, the Unit Supervisor would……

A. direct the Reactor Operator to raise RHR Train A flow by opening the RHR heat exchanger outlet flow control valve in accordance with the annunciator response procedure for Annunciator 1M2 Window F4.

B. direct the Reactor Operator to immediately start RHR Pump B per the "Skill of the Craft" guidance found in the Conduct of Operations.

C. enter 0POP04-RC-0007, Mode 5 Or Mode 6 LOCA With The Reactor Vessel Head On and direct the Reactor Operator to close all RHR Train suction isolation valves.

D. enter 0POP04-RH-0001, Loss of Residual Heat Removal, and direct the Reactor Operator to place RHR Train B in service in accordance with 0POP04-RH-0001.

Answer: D enter 0POP04-RH-0001, Loss of Residual Heat Removal, and direct the Reactor Operator to place RHR Train B in service in accordance with 0POP04-RH-0001.

Exam Bank No.: 1764 **SRO Outline Number:**

K/A Catalog Number: 005 G2.4.4 **Tier:** 2 **Group/Category:** 1

SRO Importance: 4.7 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Residual Heat Removal System: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

STP Lesson: LOT 201.09 **Objective Number:** 4245

GIVEN a plant or system condition, PREDICT the operation of the Residual Heat Removal system.

Reference: 0POP04-RH-0001, Loss of Residual Heat Removal, Rev 21

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: RHR PUMP 1A DISCH FLOW LO alarm indicates that the pump has tripped on low flow. Attempts to change flow do not work because the 1A RHR pump is tripped. Annunciator Response directs entry into 0POP04-RH-0001, Loss of Residual Heat Removal.

B: INCORRECT: Although we want to place Train B RHR in service due to the loss of RHR Pump 1A, this is not done using Skill of the Craft. Written instructions exist for operation of the RHR System.

C: INCORRECT: Per POP04-RC-0007, RHR PUMP1A DISCH FLOW LO is not an indication of a leak. Therefore entry into 0POP04-RC-0007 and closing all RHR suction valve is an inappropriate action.

D: CORRECT: RHR PUMP 1A DISCH FLOW LO annunciator response directs the operator to perform the actions of 0POP04-RH-0001, Loss of Residual Heat Removal. This alarm is also an entry condition for 0POP04-RH-0001. This off-normal procedure directs the operator to place a standby train of RHR in service.

Question Level: H **Question Difficulty** 3

Justification:

Candidate must determine that the Annunciator indicates that the RHR Pump has tripped and must then select the appropriate procedure for placing a standby train of RHR in service.

Exam Bank No.: 1765 **Last used on an NRC exam:**

Due to system conditions on Unit 1, the Instrument Air Dryers are plugging up and Instrument Air pressure is currently 95 psig.

Which of the following correctly describes the impact on the Instrument Air System and the actions the Unit Supervisor should take?

A. Instrument air pressure will continue to lower indefinitely. Enter POP02-IA-0003, Instrument Air System Operation, and place instrument air compressor #14 in service locally.

B. Instrument air pressure will continue to lower until Dryer Bypass Valve PV-9983 automatically opens. Enter POP02-IA-0003, Instrument Air System Operation, and place instrument air compressor #14 in service locally.

C. Instrument air pressure will continue to lower indefinitely. Enter POP04-IA-0001, Loss of Instrument Air, and verify Unit 2 is supplying instrument air to the yard systems.

D. Instrument air pressure will continue to lower until Dryer Bypass Valve PV-9983 automatically opens. Enter POP04-IA-0001, Loss of Instrument Air, and verify Unit 2 is supplying instrument air to the yard systems.

Answer: D Instrument air pressure will continue to lower until Dryer Bypass Valve PV-9983 automatically opens. Enter POP04-IA-0001, Loss of Instrument Air, and verify Unit 2 is supplying instrument air to the yard systems.

Exam Bank No.: 1765 **SRO Outline Number:**

K/A Catalog Number: 078 A2.01 **Tier:** 2 **Group/Category:** 1

SRO Importance: 2.9 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to (a) predict the impacts of the following malfunctions on the IAS; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Air dryer and filter malfunctions.

STP Lesson: LOT 202.26 **Objective Number:** 25609

GIVEN a plant or system condition, PREDICT the operation of the Instrument and Service Air system.

Reference: NLO200.15 PowerPoint slide 49; POP04-IA-0001, step 4 (rev 13)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Pressure would not continually lower as stated. Although the #14 compressor can only be operated in the local mode, it too discharges upstream of the malfunctioning filters and would not help improve the situation.

B: INCORRECT: The filter assembly with an automatic bypass is the filter/driers. Although the #14 compressor can only be operated in the local mode, it too discharges upstream of the malfunctioning filters and would not help improve the situation.

C: INCORRECT: Pressure would not continually lower as stated. Procedure action is correct

D: CORRECT: The filter assembly with an automatic bypass is the filter/driers. The stated conditions would require entry into the POP04 and in these conditions, the crew would verify the opposite unit was supplying air to the yard.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must must assess the given conditions, and using system knowledge, determine the affect of the malfunction. The using knowledge of procedures, assess the conditions to determine the correct transition.

Exam Bank No.: 1766 **Last used on an NRC exam:**

The Unit is at 100% power.

An electrical malfunction results in the loss of the load center supplying the in service Spent Fuel Pool Cooling Pump, causing Annunciator 22M2-F6, SFP TROUBLE, to alarm.

Which ONE of the following correctly describes the impact of the loss AND the action to be taken by the Unit Supervisor?

With Spent Fuel Pool level greater than the Technical Specification requirements, Spent Fuel Pool Temperature ......

A. will NOT rise to the boiling point. Enter the annunciator response procedure for the SFP TROUBLE alarm and start the standby pump.

B. WILL rise to the boiling point. Enter the annunciator response procedure for the SFP TROUBLE alarm and start the standby pump.

C. will NOT rise to the boiling point. Since the loss of the load center also disabled the standby pump, enter POP04-FC-0001, Loss of Spent Fuel Pool Level or Cooling, and add water to the Spent Fuel Pool using a LHSI pump to maintain temperature.

D. WILL rise to the boiling point. Since the loss of the load center also disabled the standby pump, enter POP04-FC-0001, Loss of Spent Fuel Pool Level or Cooling, and add water to the Spent Fuel Pool using a LHSI pump to maintain temperature.

Answer: B WILL rise to the boiling point. Enter the annunciator response procedure for the SFP TROUBLE alarm and start the standby pump.

Exam Bank No.: 1766 **SRO Outline Number:**

K/A Catalog Number: 033 A2.02 **Tier:** 2 **Group/Category:** 2

SRO Importance: 3.0 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Ability to (a) predict the impacts of the following malfunctions on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Loss of SFPCS

STP Lesson: LOT 201.42 **Objective Number:** 92051

GIVEN a plant or system condition, PREDICT the operation of the Spent Fuel Pool Cooling and Cleanup System.

Reference: Tech Spec 3.9.11.1 bases; 0POP09-AN-22M2 (Rev 21), window F-6

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Requirements for water level in the SFP are for dose considerations, not mass considerations for cooling (temperature will rise to boiling). The annunciator response for SFP Trouble directs the operators to start the standby pump upon trip of a running pump.

B: CORRECT: Requirements for water level in the SFP are for dose considerations, not mass considerations for cooling. The annunciator response for SFP Trouble directs the operators to start the standby pump upon trip of a running pump.

C: INCORRECT: Requirements for water level in the SFP are for dose considerations, not mass considerations for cooling (temperature will rise to boiling). While the POP04 does contain directions for using a LHSI pump to maintain SFP level, it is only used if normal means are not sufficient.

D: INCORRECT: Requirements for water level in the SFP are for dose considerations, not mass considerations for cooling. While the POP04 does contain directions for using a LHSI pump to maintain SFP level, it is only used if normal means are not sufficient.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must assess plant conditions and using system and procedure knowledge, predict the impact on the plant and the action required to help mitigate the situation.

Exam Bank No.: 1767 **Last used on an NRC exam:**

The following conditions exist:

- RT-8077, 60’ MAB Area Radiation Monitor, is currently reading $6x10^3$ mR/hr.
- Health Physics has validated the reading with local measurements.
- The 24 hour average for RT-8077 is 3 mR/hr.

Which ONE of the following correctly identifies the required Emergency Classification for this condition AND the required notifications for the classification?

A. Notification of Unusual Event (UE); notify the State/County within 15 minutes and the NRC within 1 hour of declaring the event.

B. Notification of Unusual Event (UE); notify the NRC within 15 minutes and the State/County within 1 hour of declaring the event.

C. Alert; notify the State/County within 15 minutes and the NRC within 1 hour of declaring the event.

D. Alert; notify the NRC within 15 minutes and the State/County within 1 hour of declaring the event.

Answer: C Alert; notify the State/County within 15 minutes and the NRC within 1 hour of declaring the event.

Exam Bank No.: 1767 **SRO Outline Number:**

K/A Catalog Number: 072 G2.4.30 **Tier:** 2 **Group/Category:** 2

SRO Importance: 4.1 **10CFR Reference or SRO Objective:** 55.43(b)(4)

Area Radiation Monitoring: Knowledge of events relating to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.

STP Lesson: LOT 803.14 **Objective Number:** SRO-74026

Given an emergency condition and a copy of the emergency classification tables from 0ERP01-ZV-IN01, Emergency Classification, CLASSIFY the emergency condition.

Reference: ERP01-ZV-IN01, R8, pages 25/26; ERP01-ZV-IN02, R22, step3.1

Attached Reference ☑ **Attachment:** ERP01-ZV-IN01, pages 24-26

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Although the threshold for a UE has been exceeded, it is not the correct call since a higher threshold (Alert) has also been exceeded. Notifications listed are correct.

B: INCORRECT: Although the threshold for a UE has been exceeded, it is not the correct call since a higher threshold (Alert) has also been exceeded. Notifications listed are reversed.

C: CORRECT: Condition meets the threshold for an Alert. The required notifications are 15 minutes for the State/County and 1 hour for the NRC.

D: INCORRECT: Condition meets the threshold for an Alert. Notifications listed are reversed.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must correctly use the given references to determine the required classification and must have knowledge of the notification requirements found in the Emergency Plan.

Exam Bank No.: 1768 **Last used on an NRC exam:**

Which of the following identify requirements that must be met to be considered a Senior Reactor Operator (SRO) as part of a shift crew for the next calendar quarter in accordance with POP01-ZA-0014, Licensed Operator License Maintenance?

1. Current in Licensed Operator Requalification (LOR)
2. License physical current and valid
3. Respirator physical current and valid
4. Completed five 12-hour shifts within the current quarter
5. Completed a plant tour within the current quarter

A. 2, 3, 4 ONLY

B. 2, 4, 5 ONLY

C. 1, 2, 4, 5

D. 1, 2, 3, 4

Answer: D 1, 2, 3, 4

Exam Bank No.: 1768 **SRO Outline Number:**

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

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(1)

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

STP Lesson: LOT 507.01 **Objective Number:** 92186

Given the title of an administrative procedure, DISCUSS the requirements associated with the referenced procedure.

Reference: POP01-ZA-0014, R22, step 4.3.1

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Answer is not complete, item 1 is also required.

B: INCORRECT: Items 1 and 3 are required, item 5 is not required for license maintenance (but is required for license activation).

C: INCORRECT: Item 3 is required, item 5 is not required for license maintenance (but is required for license activation).

D: CORRECT: As required by POP01-ZA-0014

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the requirements to maintain an active license.

Exam Bank No.: 1769 **Last used on an NRC exam:**

Given the following:

- Unit 1 is at 100% power.
- E1A11 Breaker #5 tripped causing a loss of power to the Train R Reactor Trip Breaker Shunt Trip Coil

Which ONE of the following describes how the Trip Breaker will respond and what subsequent actions must be taken by the Unit Supervisor?

Train R Reactor Trip Breaker .....

A. trips OPEN. Restore the Train R Reactor Trip Breaker Shunt Trip to operable status within 24 hours or be in Hot Standby within the following 6 hours.

B. trips OPEN. Restore the Train R Reactor Trip Breaker Shunt Trip to operable status within 48 hours or ensure that both Train R and Train S Reactor Trip Breakers are open within the next hour.

C. remains CLOSED. Restore the Train R Reactor Trip Breaker Shunt Trip to operable status within 24 hours or be in Hot Standby within the following 6 hours.

D. remains CLOSED. Restore the Train R Reactor Trip Breaker Shunt Trip to operable status within 48 hours or place the unit in Hot Standby within the following 6 hours.

Answer: D Breaker remains CLOSED. Restore the Train R Reactor Trip Breaker Shunt Trip to operable status within 48 hours or place the unit in Hot Standby within the following 6 hours.

Exam Bank No.: 1769 **SRO Outline Number:**

K/A Catalog Number: 012 A2.07 **Tier:** 2 **Group/Category:** 1

SRO Importance: 3.7 **10CFR Reference or SRO Objective:** 55.43(b)(2)

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 dc control power.

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: Tech Spec 3.3.1and Table 3.3-1, LOT201.20 Solid State Protection System

Attached Reference ☑ **Attachment:** Tech Spec 3.3.1 and Table 3.3-1 (Tech Spec pages 3/4 3-1 through 3/4 3-8)

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Train R Reactor Trip Breaker will NOT trip. Incorrect Action applied.

B: INCORRECT: Train R Reactor Trip Breaker will NOT trip. If the breaker were to trip, then the applied Action would be correct. However the breaker would not trip and therefore this them makes the applied Action incorrect.

C: INCORRECT: Train R Reactor Trip Breaker would remain closed. Incorrect Action applied.

D: CORRECT: Train R Reactor Trip Breaker would remain closed. Correct action applied.

Question Level: H **Question Difficulty** 3

Justification:

Candidate must understand the impact of the loss of 125 VDC control power to the Reactor Trip Breaker Shut Trip Coil and that it requires the Shunt Trip to be energized to trip its associated Reactor Trip Breaker. The Candidate must then further determine using Technical Specifications which applicable Action Statements apply.

Exam Bank No.: 1770 **Last used on an NRC exam:**

Given the following:

- An ALERT has been declared on Unit 1.
- The Shift Manager is in the process of transferring the Emergency Director responsibilities to the Technical Support Center (TSC) Manager.

Which ONE of the following responsibilities is retained by the Shift Manager (i.e. NOT transferred)?

A. Upgrading Emergency Classifications.

B. Approving commitments to the Nuclear Regulatory Commission.

C. Approving departure from license conditions per 10CFR50.54(x) for Control Room operator actions.

D. Approving communications with the Nuclear Regulatory Commission.

Answer: C Approving departure from license conditions per 10CFR50.54(x) for Control Room Operator actions.

Exam Bank No.: 1770 **SRO Outline Number:**

K/A Catalog Number: G2.4.38 **Tier:** 3 **Group/Category:**

SRO Importance: 4.4 **10CFR Reference or SRO Objective:** Objective SRO-47030

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:** SRO-47030

DISCUSS the duties and responsibilities of the Shift Supervisor as delineated in 0ERP01-ZV-SH01, Shift Supervisor.

Reference: 0ERP01-ZV-SH01, Shift Supervisor Step 5.9.3 (Rev 22)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Upgrading Emergency Classifications is transferred from the SS to the TSC Manager.

B: INCORRECT: Approving commitments to the NRC is transferred from the SS to the TSC Manager.

C: CORRECT: Approving departure from license conditions per 10CFR50.54(x) for Control Room operator actions is retained by the SS. This duty is not transferred.

D: INCORRECT: Approving communications with the Nuclear Regulatory Commission is transferred to the TSC Manager.

Question Level: F **Question Difficulty** 3

Justification:

Candidate must know that even though Emergency Director responsibilities are transferred to the TSC or EOF, departure from license conditions per 10CFR50.54(x) for control room actions must still reside with the SS in command and control.

Exam Bank No.: 1771 **Last used on an NRC exam:**

A fire has been reported in the Unit 1 Turbine Generator Building.

Which of the following describes the action to be taken by the Unit 1 Unit Supervisor?

A. Enter PGP03-ZF-0011, STPEGS Fire Brigade, contact the Unit 2 Control Room with fire information and request their coordination of the fire response.

B. Enter PGP03-ZF-0011, STPEGS Fire Brigade, sound the fire alarm and make a plant announcement to activate the fire brigade.

C. Enter POP04-ZO-0008, Fire/Explosion, contact the Unit 2 Control Room with fire information and request their coordination of the fire response.

D. Enter POP04-ZO-0008, Fire/Explosion, sound the fire alarm and make a plant announcement to activate the fire brigade.

Answer: D Enter POP04-ZO-0008, Fire/Explosion, sound the fire alarm and make a plant announcement to activate the fire brigade.

Exam Bank No.: 1771 **SRO Outline Number:**

K/A Catalog Number: G2.4.27 **Tier:** 3 **Group/Category:** 4

SRO Importance: 3.9 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Knowledge of "fire in the plant" procedures.

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: POP04-ZO-0008, R18, Entry conditions and steps 4 and 5.

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: The referenced procedure describes the organization and duties of the fire brigade, the off-normal procedure activated it when needed. Coordination of the fire response is only transferred to the unaffected unit if the affected unit's control room is evacuated per the off-normal procedure..

B: INCORRECT: The referenced procedure describes the organization and duties of the fire brigade, the off-normal procedure activated it when needed. The actions listed are correct per the off-normal procedure.

C: INCORRECT: Correct procedure, however coordination of the fire response is only transferred to the unaffected unit if the affected unit's control room is evacuated.

D: CORRECT: A reported fire is an entry condition for POP04-ZO-0008 which goes on to sound the alarm and make announcements to activiate the fire brigade.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the procedures used to activate the response of the fire brigade.

Exam Bank No.: 1772 **Last used on an NRC exam:**

Given the following conditions:

- Unit 2 is at 100% power
- An Unusual Event has been declared due to high RT-8010B, Unit Vent Stack, radioactivity levels caused by an ongoing water leak in the MAB.
- The on-duty Shift Manager is the Emergency Director.

In accordance with PGP03-ZA-0090, Work Process Program, which ONE of the following describes a method which could be implemented to expeditiously repair the leak and stop the release?

Maintenance can be directed to begin work without an approved work package after....

A. The Shift Manager declares the repair Emergent Work.

B. The Shift Manager declares the repair Emergency Maintenance.

C. The Plant Manager declares the repair Emergent Work.

D. The Plant Manager declares the repair Emergency Maintenance.

Answer: B The Shift Supervisor declares the repair Emergency Maintenance.

Exam Bank No.: 1772 **SRO Outline Number:**

K/A Catalog Number: G2.2.17 **Tier:** 3 **Group/Category:** 2

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(4)

Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.

STP Lesson: LOT 802.33 **Objective Number:** SRO-50014

DESCRIBE the Work Process Priority classifications for condition reports.

Reference: PGP03-ZA-0090, R32, Step 2.8

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Emergent work requires a work package.

B: CORRECT: In accordance with site procedures

C: INCORRECT: Emergent work requires a work package.

D: INCORRECT: The Shift Supervisor/Emergency Director only has the authority to declare emergency maintenance.

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have a knowledge of the requirements for emergency maintenance under the work process program.

Exam Bank No.: 1774 **Last used on an NRC exam:**

The Unit has just entered Mode 3 after operating for 6 months.

Which ONE of the following conditions indicates a loss of containment integrity and would require entry into the action statement for Technical Specification 3.6.1.1, Containment Integrity? (consider each condition separately)

A. Manual containment isolation valves located inside containment have not been verified since the last time the Unit was in Mode 5.

B. Work is in progress on a motor operated containment isolation valve's breaker with the valve de-energized in the closed position.

C. The inner personnel airlock door was declared inoperable 30 minutes ago. The outer door has just been opened to allow entry of maintenance personnel to work on the inner door.

D. It was identified 72 hours ago that the personnel airlock exceeded Surveillance 4.6.1.3.a leakage rate requirements (overall leakage requirements of 3.6.1.2, Containment Leakage are still met).

Answer: D It was identified 72 hours ago that the personnel airlock exceeded Surveillance 4.6.1.3.a leakage rate requirements (overall leakage requirements of 3.6.1.2, Containment Leakage are still met).

Exam Bank No.: 1774 **SRO Outline Number:**

K/A Catalog Number: APE069 AA2.01 **Tier:** 1 **Group/Category:** 2

SRO Importance: 4.3 **10CFR Reference or SRO Objective:** 55.43(b)(2)

Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Loss of containment integrity.

STP Lesson: LOT 503.01 **Objective Number:** 80056

Given a system scenario, DETERMINE the applicable Technical Specification and/or the Technical Requirements Manual (TRM) for the system and APPLY the specification(s).

Reference: Tech Spec 3.6.1.1 and 3.6.1.3

Attached Reference ☑ **Attachment:** Tech Spec 3.6.1.1 and 3.6.1.3 (pages 3/4 6-1, 3/4 6-5 and 3/4 6-6)

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Per the footnote on TS 3.6.1.1, valves inside containment need only be verified each time the Unit enters Mode 5 unless it has been less than 92 days since the last verification was completed.

B: INCORRECT: Per surveillance requirement 4.6.1.1.a, a penetration with a closed, deactivated automatic valve is acceptable.

C: INCORRECT: Per TS 3.6.1.3, action A, the operable (outer) door must be closed with 1 hour. As such the stated condition does not surpass that time limit.

D: CORRECT: Per TS 3.6.1.3, the unit should have been place in Mode 5 within 54 hours. Since it was not, then the surveillance requirements of TS 3.6.1.1 are not met and the action for TS 3.6.1.1 would need to be implemented.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must apply the given conditions to the Technical Specification requirements to determine if requirements are met.

Exam Bank No.: 1794 **Last used on an NRC exam:**

Given the following conditions:

- Unit 1 is in a refueling outage with core reload in progress.
- Extended Range Channel NI-45 is out of service.
- Source Range Channel NI-31 is providing audible count rate indication.
- Source Range Channel NI-32 is NOT capable of providing audible count rate indication.
- Source Range Channel NI-31 has a significantly lower detector voltage than Source Range Channel NI-32.

Which one of the below correctly describes the effect on Source Range Channel NI-31 due to the lower detector voltage and the applicable Technical Specification requirements if NI-31 were declared inoperable?

With a lower detector voltage on NI-31, the indicated count rate should be …

A. LOWER than NI-32. Technical Specifications for Source Range Flux Monitoring would still be met if NI-31 was declared inoperable. Core Alterations could continue.

B. LOWER than NI-32. Technical Specifications for Source Range Flux Monitoring would NOT be met if NI-31 was declared inoperable. Core Alterations would be immediately suspended.

C. HIGHER than NI-32. Technical Specifications for Source Range Flux Monitoring would still be met if NI-31 was declared inoperable. Core Alterations could continue.

D. HIGHER than NI-32. Technical Specifications for Source Range Flux Monitoring would NOT be met if NI-31 was declared inoperable. Core Alterations would be immediately suspended.

Answer: B lower than NI-32, Technical Specifications for Source Range Flux Monitoring would NOT be met if NI-31 was declared inoperable. Core Alterations would be immediately suspended.

Exam Bank No.: 1794 **SRO Outline Number:**

K/A Catalog Number: APE 032 AA2.09 **Tier:** 1 **Group/Category:** 2

SRO Importance: 2.9 **10CFR Reference or SRO Objective:** 55.43(b)(6)

Loss of Source Range NI: Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Effect of improper HV setting.

STP Lesson: LOT 201.16 **Objective Number:** 92102

Given the topic or title of a specification included in the Technical Specifications, or the Technical Requirements Manual (TRM), DESCRIBE the general requirements of the specification to include components or administrative requirements affected, limitations, major time frames involved, major surveillance in order to comply, and the bases for the specification.

Reference: TS 3.9.2

Attached Reference ☑ **Attachment:** Tech Spec 3.9.2, Refueling Operations; Instrumentation. Pg. 3/4 9-2

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: it is correct that indicated count rate would be lower, but it is incorrect that TS Source Range Flux Monitoring requirements are met because neither of the two remaining channels (NI-32 and NI-45) can provide audilble count rate indication.

B: CORRECT: With lower detector voltage, indicated count rate will be lower. At least one of the two required Source Range Flux Monitors must be capable of providing audible count rate indication. Since NI-32 cannot and NI-45 isn't designed to give audible count rate, if NI-31 is declared inoperable, there will be no audible count rate indication. This would mean the Source Range Flux Monitoring requirements of TS 3.9.2 would not be met and Core Alterations would have to be stopped.

C: INCORRECT: it is incorrect that indicated count rate would be higher with a lower detector voltage. Additionally, it is incorrect that TS Source Range Flux Monitoring requirements are met because niether of the two remaining channels (NI-32 and NI-45) can provide audilble count rate indication.

D: INCORRECT: it is incorrect that indicated count rate would be higher with a lower detector voltage. It is correct that TS requirements are NOT met as described in the justification for correct answer 'C'.

Question Level: H **Question Difficulty** 3

Justification:

must understand how detector voltage affects Source Range Detector output and be able to apply TS requirements for Mode 6 Source Range Flux Monitoring to the given information.

Exam Bank No.: 1845 **Last used on an NRC exam:**

Given the following:

- Unit 1 was operating at 100% reactor power
- Tripping of the Main Generator Field Breaker has resulted in a Turbine trip
- The crew has completed all actions in 0POP05-EO-EO00, Reactor Trip or Safety Injection, and have transitioned to 0POP05-EO-ES01, Reactor Trip Response

The Secondary Operator reports the following alarms and associated equipment status:

- 07M3-E-7, MAIN COND VACUUM LO, condenser vacuum is 21.0 in. hg and stable
- 03M3-A-8, DG 13 TRBL, ESF Diesel Generator 13 is not running
- 06M4 - F-2, FWIV FV-7143 N2 PRESS LO, Steam Generator 1C feedwater isolation valve FV-7143 is closed

What additional actions should be taken while continuing with 0POP05-EO-ES01, Reactor Trip Response?

A. Enter 0POP04-AE-0001, First Response To Loss Of Any Or All 13.8 KV Or 4.16 KV Bus while continuing with 0POP05-EO-ES01.

B. Close all Main Steam Isolation Valves and Main Steam Isolation Bypass Valves.

C. Close ONLY the Main Steam Isolation Valve and Main Steam Isolation Bypass Valve for the 1C steam generator.

D. No actions required outside of 0POP05-EO-ES01. These alarms are expected after a Reactor Trip.

Answer: B Close all main steam isolation valves and main steam isolation bypass valves

Exam Bank No.: 1845 **SRO Outline Number:**

K/A Catalog Number: EPE 007 G2.4.45 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.3 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Reactor Trip: Ability to prioritize and interpret the significance of each annunciator or alarm

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-ES01(Rev 24); 0POP09-AN-03M3, A-8 (Rev 23); 0POP09-AN-06M4 (Rev 29); 0POP09-AN-07M (Rev 45)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Bus E1C is energized by standby bus and the EDG should not be running because no safety injection signal exists. Multiple local alarms can cause this alarm. No reason to enter 0POP04-AE-0001

B: CORRECT: Steam Dump Block Permissive actuates at 22 in. hg, therefore at 21 in. hg the steam dumps are blocked. Therefore, close MSIV's and MSIB's in accordance with the MSIV and MSIB Closure Criteria of the Conditional Information Page contained in 0POP05-EO-ES01.

C: INCORRECT: No guidance in 0POP05-EO-ES01 to close individual MSIV's . Also the applicant should understand that a feedwater isolation is normal for a reactor trip and therefore the 1C FWIV should normally be closed.

D: INCORRECT: Actions are required for main condenser low vacuum

Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze the given conditions and then based on system knowledge, prioritize the given alarms.

Exam Bank No.: 1847 **Last used on an NRC exam:**

Given the following conditions:

- Unit 1 is at 100% power
- All Pressurizer level channels indicate 57% and STABLE.
- VCT level instruments LT-112 and LT-113 indicate 27% and trending down.
- The Primary Operator reports the RC MU CONT SYS ON red light is not lit (bulb is good).

Which one of the following describes the action required?

A. No action required. The RC MU CONT SYS ON red light will illuminate during automatic makeup that normally occurs at 15% VCT level.

B. No action required. The RC MU CONT SYS ON red light will only illuminate if an alarm condition occurs during automatic makeup to the VCT.

C. Initiate makeup to the VCT utilizing the ‘Manual Makeup to RCS’ section of 0POP02-CV-0002, Makeup to the Reactor Coolant System.

D. Swap Charging Pump suction from the VCT to the RWST using the Annunciator Response Procedure for 4M08-E-2, VCT LEVEL HI/LO.

Answer: C Initiate makeup to the VCT utilizing the ‘Manual Makeup to RCS’ section of 0POP02-CV-0002, Makeup to the Reactor Coolant System.

Exam Bank No.: 1847 **SRO Outline Number:**

K/A Catalog Number: APE 022 AA2.03 **Tier:** 1 **Group/Category:** 1

SRO Importance: 3.6 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Reactor Coolant Makeup: Ability to determine and interpret the following as they apply to the Loss of Reactor Coolant Makeup: failures of flow control valve or controller

STP Lesson: **Objective Number:**

Reference: LOT201.07, Reactor Makeup Control System (Rev 12); 0POP02-CV-0001, Makeup to the Reactor Coolant System (Rev 31)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Auto make-up is inhibited with the RC MU CONT SYS ON dark. With the makeup system in a normal alignment, the red light should be on

B: INCORRECT: Auto make-up is inhibited with the RC MU CONT SYS ON dark. Also, 15% is incorrect value for auto make-up

C: CORRECT: A problem exists with the automatic makeup system since the red light is dark. The operators should attempt manual makeup.

D: INCORRECT: This annunciator will not alarm until level lowers to 15%, and only addresses swapping the charging pump suction when VCT level is les than 3%.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze the given conditions using system knowledge and determine the proper action based on procedure knowledge.

<u>Exam Bank No.:</u> 1848 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in mode 5, maintaining RCS mid-loop conditions
- RCS Level is being maintained between +11 and +6 inches by narrow range hot leg level indication (33’ 2” to 32’ 9”)
- RHR trains “A” & “B” are in service with flow throttled to 2800 gpm for each train
- Reactor Vessel Head detensioning is in progress

The Primary Operator reports the following:

- 01M2-F-1, RC MID LOOP LVL LO-LO, alarms
- RCS level indicates +0 inches (NR Hot Leg Indication) and stable
- Both in service RHR Pumps are exhibiting erratic flow and current indications

Which one of the following describes the appropriate action to take by the Unit Supervisor?

A. Enter 0POP04-RH-0001, Loss of Residual Heat Removal; STOP both RHR pumps.

B. Enter 0POP04-RH-0001, Loss of Residual Heat Removal; STOP 1 RHR pump.

C. Enter 0POP04-RC-0006, Shutdown LOCA; STOP both RHR pumps.

D. Enter 0POP04-RC-0006, Shutdown LOCA; STOP 1 RHR pump.

Answer: A Enter 0POP04-RH-0001, Loss of Residual Heat Removal; STOP both RHR pumps.

Exam Bank No.: 1848 **SRO Outline Number:**

K/A Catalog Number: APE 025 G2.1.23 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.4 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Residual Heat Removal System (RHRS): Ability to perform specific system and integrated plant procedures during all modes of plant operation

STP Lesson: **Objective Number:** 91908

Reference: 0POP03-ZG-0009 (Rev 47); 0POP04-RH-0001 (Rev 21); 0POP09-AN-01M2 (Rev 17); 0POP04-RC-0006 (Rev 15)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: RC MID LOOP LVL LO-LO alarm directs entry into 0POP04-RH-0001 if running RHR pump(s) exhibit signs of air entrainment (step 5). Step 3b of 0POP04-RH-0001 stops all RHR pumps if NR Hot Leg level is < 1 inch.

B: INCORRECT: Step 3b of 0POP04-RH-0001 stops all RHR pumps if NR Hot Leg level is < 1 inch (does not leave 1 pump running)

C: INCORRECT: 0POP04-RC-0006, Shutdown LOCA, is only applicable in Mode 3 with RCS pressure less than 1000 psig or Mode 4.

D: INCORRECT: 0POP04-RC-0006, Shutdown LOCA, is only applicable in Mode 3 with RCS pressure less than 1000 psig or Mode 4

Question Level: H **Question Difficulty** 3

Justification:

The candidate must use given conditions and knowledge of procedure to determine correct procedure to enter and knowledge of the procdure itself to determine the correct action.

Exam Bank No.: 1849 **Last used on an NRC exam:**

Given the following conditions:

- Unit 1 was operating at 100% power when a feedwater line break occurred inside containment.
- Steam Generator narrow range levels are as follows:
  - SG 1A = 7%
  - SG 1B = 8%
  - SG 1C = 7%
  - SG 1D = 0%
- No Steam Generator feedwater (main or auxiliary) is available. Attempts are being made to restore AFW pumps.
- Containment pressure is 6 psig and stable

Which one of the below describes the existing emergency classification and the MINIMUM additional barrier degradation required for escalation to the next higher classification?

A. The existing emergency classification is an ALERT. Escalation criteria would be Loss of Fuel Clad OR Loss of RCS

B. The existing emergency classification is an ALERT. Escalation criteria would be Loss of Containment

C. The existing emergency classification is a SITE AREA EMERGENCY. Escalation criteria would be Loss of Fuel Clad AND Loss of Containment

D. The existing emergency classification is a SITE AREA EMERGENCY. Escalation criteria would be Loss of RCS AND Potential Loss of Containment

Answer: C The existing emergency classification is a SITE AREA EMERGENCY. Escalation criteria would be Loss of Fuel Clad AND Loss of Containment

Exam Bank No.: 1849 **SRO Outline Number:**

K/A Catalog Number: APE 054 G2.4.41 **Tier:** 1 **Group/Category:** 1

SRO Importance: 4.6 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Residual Heat Removal System (RHRS): Ability to perform specific system and integrated plant procedures during all modes of plant operation

STP Lesson: **Objective Number:**

Reference: 0ERP01-ZV-IN01, Emergency Classification (Rev 8); 0POP-EO-F003, Heat Sink Critical Safety Function Status Tree (Rev 6)

Attached Reference ☑ **Attachment:** 0ERP01-ZV-IN01, pages 10 and 11

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT - This event is classified as a Site Area Emergency

B: INCORRECT - This event is classified as a Site Area Emergency

C: CORRECT - With no feedwater available and all SG's less than 34% (containment is adverse), the critical safety function for Heat Sink is RED. The Fission Product Barrier Degradation Initiating Condition Matrix dictates a Potential Loss of Fuel Clad (3) AND a Potential Loss of RCS (3) for Heat Sink being RED. This equates to a SAE (6) for potential loss of both fuel clad and RCS. Escalation to a General Emergency would require loss of ANY two barriers AND potential loss of a third barrier; or a loss of three barriers. This answer, a potential loss of RCS (due to Heat Sink RED) along with Loss of Fuel Clad and Loss of Containment would fulfill the requirements for a General Emergency.

D: INCORRECT - Classification is correct, however the escalation criteria is incorrect. This answer has a loss of one barrier (RCS) and a potential loss of two barriers (Fuel Clad due to Red Heat Sink and Containment) and therefore does not meet the General Emergency criteria.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must be able to apply the given conditions to the given reference to obtain the correct response.

Exam Bank No.: 1850 **Last used on an NRC exam:**

Given the following Unit 2 conditions:

- A loss of all AC power has occurred.
- The Control Room operators are monitoring DC bus voltages in accordance with Addendum 4, Vital DC Bus Monitoring, of 0POP05-EO-EC00, Loss of All AC Power.
- Train “A” main battery breaker E2A11 has been opened due to low bus voltage.

Subsequently, the Primary Operator reports the following:

- Train B Class 1E battery voltage is 105 VDC
- Train C Class 1E battery voltage is 110 VDC

Which one of the following identifies the REQUIRED action to be taken AND the basis for the action?

A. Open Train “B” main battery breaker E2B11 to conserve the battery should ESF DG 22 become available.

B. Open BOTH Train “B” and Train “C” main battery breakers (E2B11 and E2C11) to conserve the batteries should their respective ESF DG’s (DG 22 and DG 23) become available.

C. Open Train “B” main battery breaker E2B11 to prevent buildup of H_2 gas to explosive concentrations due to the excessive discharge with no ventilation.

D. Open BOTH Train “B” and Train “C” main battery breakers (E2B11 and E2C11) to prevent buildup of H_2 gas to explosive concentrations due to the excessive discharge with no ventilation.

Answer: A Open train B main battery breaker E2B11 to conserve the battery should EDG 22 become available

Exam Bank No.: 1850 **SRO Outline Number:**

K/A Catalog Number: EPE 055 EA2.05 **Tier:** 1 **Group/Category:** 1

SRO Importance: 3.7 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Loss of Offsite and Onsite Power (Station Blackout): Ability to determine or interpret the following as they apply to a Station Blackout: When battery is approaching fully discharged

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-EC00, Loss of All AC Power (Rev 19); LOT504.22, Loss of All AC Power (Rev 10)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: A note prior to step 4 of Addendum 4, Vital DC Bus Monitoring, states that battery output breakers should be opened if their respective DC voltage lowers to LESS THAN OR EQUAL TO 105.5 VDC in order to conserve the battery should a STBY DG become available

B: INCORRECT: Train C voltage is above 105.5 VDC and therefore the battery breaker should not be opened at this time

C: INCORRECT: Wrong basis. A caution before step 4 of Addendum states: Do not allow battery voltages to drop to LESS THAN 105 VDC for plant equipment protection. No battery voltages are BELOW 105 VDC

D: INCORRECT: same as C

Question Level: F **Question Difficulty** 3

Justification:

The candidate must have knowledge of precedural requirements for battery monitoring.

Exam Bank No.: 1851 **Last used on an NRC exam:**

Given the following:

- A grid disturbance has occurred.
- The Secondary Operator reports Main Generator output to be 1350 MW and Main Generator reactive load to be 100 MVARS IN.

Which one of the following describes the condition of the Main Generator AND any action to be taken by the Unit Supervisor?

The Main Generator is operating...

A. within the limits of the Main Generator Capability Curve; no further action is required.

B. outside the limits of the Main Generator Capability Curve; raise hydrogen gas pressure in accordance with 0POP02-GG-0001, Generator Hydrogen and Carbon Dioxide Gas System, to establish operation within limits.

C. outside the limits of the Main Generator Capability Curve; raise Main Generator excitation in accordance with 0POP03-ZG-0008, Power Operations, to establish operation within limits.

D. outside the limits of the Main Generator Capability Curve; lower Main Generator excitation in accordance with 0POP03-ZG-0008, Power Operations, to establish operation within limits.

Answer: C outside the limits of the Main Generator Capability Curve; raise Main Generator excitation in accordance with 0POP03-ZG-0008, Power Operations, to establish operation within limits.

Exam Bank No.: 1851 **SRO Outline Number:**

K/A Catalog Number: APE 077 AA2.10 **Tier:** 1 **Group/Category:** 1

SRO Importance: 3.8 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Generator Voltage and Electric Grid Disturbances: Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Generator overheating and the required actions

STP Lesson: **Objective Number:**

Reference: Generator Capability Curve; 0POP03-ZG-0008, Power Operations (Rev 48)

Attached Reference ☑ **Attachment:** Main Generator Capability Curve (PCB 107.01)

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Given conditions place operation of the generator outside the limits of the capability curve

B: INCORRECT: Operation is outside the curve; even though high temperature is the concern, lowering cold gas temperature will not correct the condition

C: CORRECT: Operation is outside the curve and raising excitation will bring operation back within limits of the curve which is designed to prevent overheating of the generator

D: INCORRECT: Operation is outside the curve; lowering excitation will make the condition worse

Question Level: H **Question Difficulty** 3

Justification:

The candidate must be able to apply the given conditions to the curve and based on this information, determine the corrective action using integrated plant knowledge.

Exam Bank No.: 1852 **Last used on an NRC exam:**

Given the following:

- Unit 1 is in Mode 5 following refueling
- The Reactor Coolant System has been filled and vented

In accordance with 0POP02-CV-0001, Makeup to the Reactor Coolant System, which one of the following describes the required position and bases for CV-0198, “RX MAKEUP WATER ISOL”?

CV-0198 will be ...

A. full open to allow normal use of the Reactor Makeup Control System.

B. locked in place to limit flow to 200 gpm in order to satisfy 0PGP03-ZO-0042, Reactivity Management Program, requirements.

C. locked in place to limit flow to 110 gpm in order to satisfy Technical Specification requirements.

D. locked closed in order to satisfy Technical Specification requirements.

Answer: C locked in place to limit flow to 110 gpm in order to satisfy Technical Specification requirements.

Exam Bank No.: 1852 **SRO Outline Number:**

K/A Catalog Number: 004 G2.1.32 **Tier:** 2 **Group/Category:** 1

SRO Importance: 4.0 **10CFR Reference or SRO Objective:** 55.43(b)(2)

Chemical and Volume Control System: Ability to explain and apply system limits and precautions

STP Lesson: **Objective Number:**

Reference: 0POP02-CV-0001, Makeup to the Reactor Coolant System (Rev 31)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: This is the normal position in Mode 1-3 which is not correct for this condition.

B: INCORRECT: The valve should be locked in place, however the referenced flow rate is not correct.

C: CORRECT: The valve is locked in this position as required by Tech Specs

D: INCORRECT: This is the Tech Spec required position when the RCS loops are not filled.

Question Level: F **Question Difficulty** 3

Justification:

Using the given conditions, the candidate must apply the correct Tech Spec requirement.

Exam Bank No.: 1853 **Last used on an NRC exam:**

Given the following for Unit 1:

- Plant heatup is in progress per 0POP03-ZG-0001, Plant Heatup
- RCS Tave is 355 °F
- A loss of ALL Instrument Air has occurred.

Which one of the below correctly describes (1) the effect on the CCW Train RHR Heat Exchanger Outlet Valves (CCW OUTL FV-4531/4548/4565) and (2) the procedural guidance to mitigate the consequences for the operating CCW Train/s?

A. Fails OPEN; Align CCW Trains in accordance with 0POP04-RH-0001, Loss of Residual Heat Removal.

B. Fails OPEN; Align CCW Trains in accordance with the Annunciator Response Procedures for the CCW HX OUTLET FLOW HI/LO alarms.

C. Fails CLOSED; Align CCW Trains in accordance with the Annunciator Response Procedures for the CCW HX OUTLET FLOW HI/LO alarms.

D. Fails CLOSED; Align CCW Trains in accordance with 0POP04-RH-0001, Loss of Residual Heat Removal.

Answer: B Fails OPEN; Align CCW Trains in accordance with the Annunciator Response Procedures for the CCW HX OUTLET FLOW HI/LO alarms.

Exam Bank No.: 1853 **SRO Outline Number:**

K/A Catalog Number: 008 A2.05 **Tier:** 2 **Group/Category:** 1

SRO Importance: 3.5 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Component Cooling Water System: Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Effect of loss of instrument and control air on the position of the CCW valves that are air operated

STP Lesson: **Objective Number:** 5213

Reference: 0POP04-IA-0001, Loss Of Instrument Air (Rev 13); LOT201.12, Component Cooling Water System (Rev 13)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Valves do fail open, however POP04-RH-0001 is not the correct procedure

B: CORRECT: Correct failure position and procedure

C: INCORRECT: Valves fail open, correct procedure.

D: INCORRECT: Valves fail open, incorrect procedure

Question Level: H **Question Difficulty** 3

Justification:

The candidate must determine the failure mode based on given conditions and determine the corrrect procedure response using the given conditions.

Exam Bank No.: 1854 **Last used on an NRC exam:**

Given the following:

- A Unit 1 Reactor trip occurred due to Loss of Offsite Power (LOOP).
- Control Room operators are currently performing the actions of 0POP05-EO-ES01, Reactor Trip Response.
- Current plant conditions are as follows:
  - RCS Tave - 571° F and stable
  - Pressurizer level – 27% and stable
  - Pressurizer pressure – 2245 psig and stable

The Primary Operator initiates Auxiliary Spray to lower Pressurizer pressure by performing the following:

- Opens AUX SPRY Valve, PV-3119
- Closes LOOP ISOL MOV-0003
- Manually opens Charging Flow Control Valve, FCV-0205, to maximize spray flow and reduce Pressurizer pressure to 2235 psig.

When RCS pressure reaches 2235 psig, the operator attempts to close the Aux Spray Valve, but the spray valve FAILS TO CLOSE.

Assuming no further operator action, which one of the following describes the Pressurizer LEVEL response and the correct procedure usage to mitigate the failure?

Pressurizer level will…

A. INCREASE; Implement 0POP04-RP-0001, Loss of Automatic Pressurizer Pressure Control.

B. INCREASE; Implement 0POP04-RP-0002, Loss of Automatic Pressurizer Level Control.

C. REMAIN STABLE; Implement 0POP04-RP-0001, Loss of Automatic Pressurizer Pressure Control.

D. REMAIN STABLE; Implement 0POP04-RP-0002, Loss of Automatic Pressurizer Level Control.

Answer: A INCREASE; Implement 0POP04-RP-0001, Loss of Automatic Pressurizer Pressure Control.

Exam Bank No.: 1854 **SRO Outline Number:**

K/A Catalog Number: 011 A2.12 **Tier:** 2 **Group/Category:** 2

SRO Importance: 3.3 **10CFR Reference or SRO Objective:** 55.43(b)(5)

Pressurizer Level Control System (PZR LCS): Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Operation of auxiliary spray

STP Lesson: **Objective Number:**

Reference: 0POP05-EO-ES01(Rev 24); 0POP04-RP-0001(Rev 14); 0POP04-RP-0002 (Rev19)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: CORRECT: Pressurizer level will increase due to lowering pressure. The Loss of Automatic Pressurizer Pressure Control procedure (step 5) will direct the operators to secure letdown and charging to mitigate the failure

B: INCORRECT: The Loss of Automatic Pressurizer Level Control procedure does not address the auxiliary spray valve

C: INCORRECT: Pressurizer level will increase, not remain stable.

D: INCORRECT: Pressurizer level will increase, not remain stable.

Question Level: H **Question Difficulty** 3

Justification:

The candidate must understand system response when saturation conditions are reached in the head and apply the given conditions to procedure entry requirements.

Exam Bank No.: 1855 **Last used on an NRC exam:**

Given the following:

- Unit 2 is at 48% power
- HIGH alarms are received on the following Radiation Monitors:
  - o RT-8027, Condenser Air Removal System
  - o RT-8043, Steam Generator Blowdown System
- Charging flow is unexpectedly rising above letdown flow

Which one of the following describes the action to be taken by the Unit Supervisor?

A. Enter 0POP04-RC-0003, Excessive RCS Leakage; bypass the Condensate Polishers to prevent contamination of the resin.

B. Enter 0POP04-RC-0003, Excessive RCS Leakage; align Steam Generator Blowdown to the Blowdown Demineralizers to minimize the spread of contamination in the secondary system.

C. Enter 0POP04-RC-0004, Steam Generator Tube Leakage; bypass the Condensate Polishers to prevent contamination of the resin.

D. Enter 0POP04-RC-0004, Steam Generator Tube Leakage; align Steam Generator Blowdown to the Blowdown Demineralizers to minimize the spread of contamination in the secondary system.

Answer: D Enter 0POP04-RC-0004, Steam Generator Tube Leakage; align Steam Generator Blowdown to the Blowdown Demineralizers to minimize the spread of contamination in the secondary system.

Exam Bank No.: 1855 **SRO Outline Number:**

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: **Objective Number:** 37872

Reference: 0POP04-RC-0004, Steam Generator Tube Leakage (Rev 23)

Attached Reference ☐ **Attachment:**

NRC Reference Req'd ☐ **Attachment:**

Source: New **Modified From**

Distractor Justification

A: INCORRECT: Based on the given conditions, POP04-RC-0004 should be entered
B: INCORRECT: Based on the given conditions, POP04-RC-0004 should be entered
C: INCORRECT: Correct procedure, incorrect action
D: CORRECT: Correct procedure, correct action

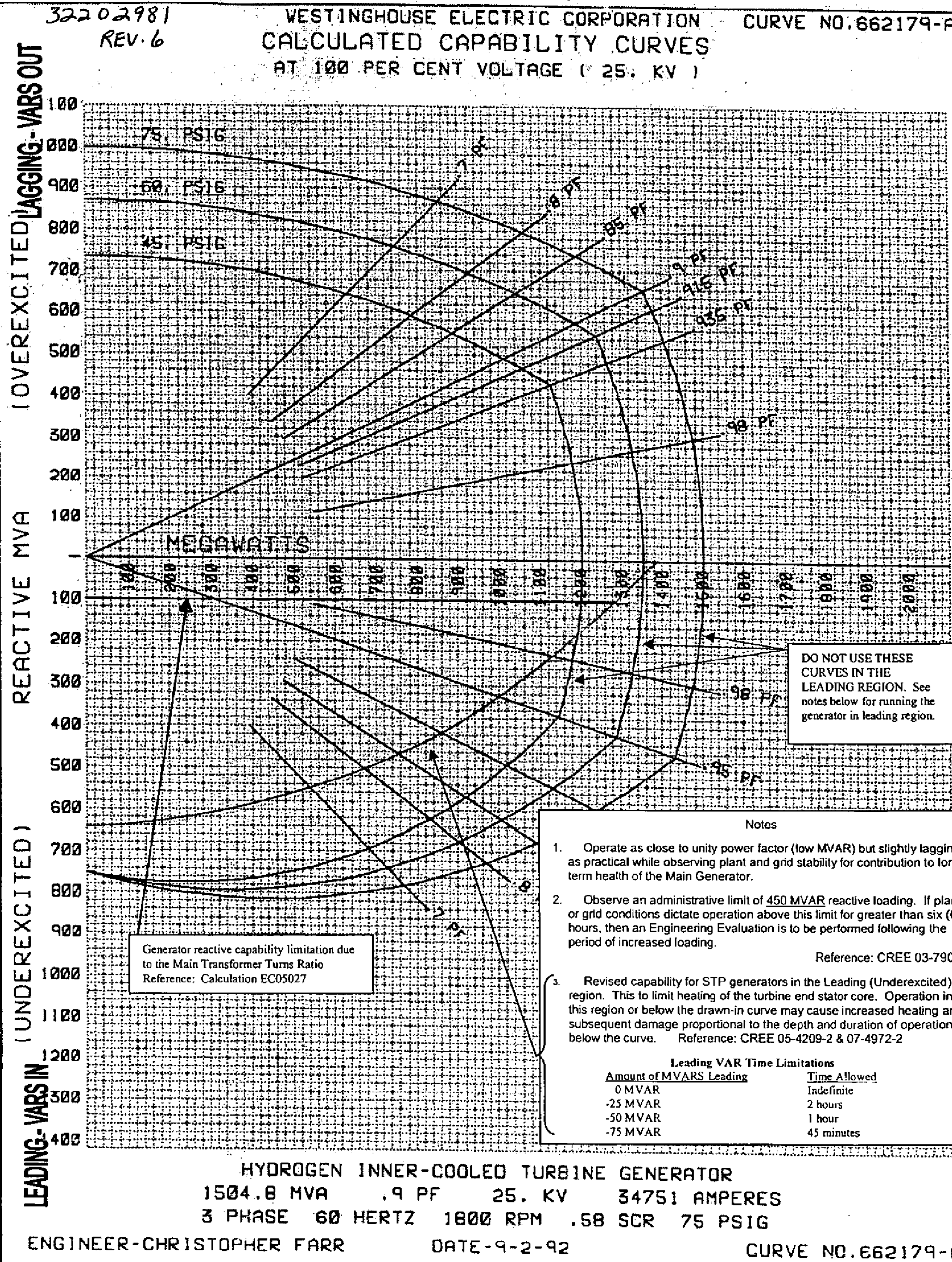
Question Level: H **Question Difficulty** 3

Justification:

The candidate must analyze the given conditins to determine the malfunction in progress and then with procedure knowledge, determine the correct entry and action.

LOT 17 NRC Exam
SRO
Reference
Package

FIGURE 7.1 MAIN GENERATOR CAPABILITY CURVE (UNIT 1)



Notes

1. Operate as close to unity power factor (low MVAR) but slightly lagging as practical while observing plant and grid stability for contribution to long-term health of the Main Generator.
2. Observe an administrative limit of 450 MVAR reactive loading. If plant or grid conditions dictate operation above this limit for greater than six (6) hours, then an Engineering Evaluation is to be performed following the period of increased loading.

   Reference: CREE 03-7907-2
3. Revised capability for STP generators in the Leading (Underexcited) region. This to limit heating of the turbine end stator core. Operation in this region or below the drawn-in curve may cause increased heating and subsequent damage proportional to the depth and duration of operation below the curve. Reference: CREE 05-4209-2 & 07-4972-2

Leading VAR Time Limitations

Amount of MVARS Leading	Time Allowed
0 MVAR	Indefinite
-25 MVAR	2 hours
-50 MVAR	1 hour
-75 MVAR	45 minutes

Prepared by: [signature] Date: 8/14/07 Reviewed by: [signature] Date: 8/15/07
Reference: VTD –W120-0014 Pg. 034 Approved by: [signature] Date: 8/28/07

RECOGNITION CATEGORY F
<u>FISSION PRODUCT BARRIER DEGRADATION</u>
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

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.

RECOGNITION CATEGORY R
RADIOLOGICAL
INITIATING CONDITION MATRIX

RADIATION LEVELS

INITIATING CONDITION	EMERGENCY ACTION LEVEL	CLASS
RG2 Unexpected Increase in Containment Radiation Levels. **Modes: 1-4**	**EAL-1** Valid reading on RCB High Range Monitors, RT-8050 or RT-8051 greater than 1,000 R/hr. **OR** Valid reading on Hatch Monitor greater than 2,222 mR/hr.	**GE**
RS2 Unexpected Increase in Containment Radiation Levels. **Modes: 1-4**	**EAL-1** Valid reading on RCB High Range Monitors, RT-8050 or RT-8051 greater than 100 R/hr. **OR** Valid reading on Hatch Monitor greater than 222 mR/hr.	**SAE**
RA2 Major Damage to Irradiated Fuel or Loss of Water Level that has or Will Result in the Uncovering of Irradiated Fuel Outside the Reactor Vessel. **Modes: At all times**	**EAL-1** Valid readings on one or more of the following monitors: FHB Exhaust, RT-8035 > 5.00 E-2 µCi/ml FHB Exhaust, RT-8036 > 5.00 E-2 µCi/ml Area Monitor (68' FHB), RT-8090 > 5,000 mR/hr **EAL-2** Valid readings on the following radiation monitor with spent nuclear fuel in the Refueling Cavity or Inside Containment Storage Area (ICSA). (Mode 6 Only). Area Monitor (68' RCB), RT-8099 > 5,000 mR/hr **EAL-3** Irradiated fuel uncovered (actual or potential) based on observation OR water level below top of fuel storage racks.	**ALERT**

RECOGNITION CATEGORY R
RADIOLOGICAL
INITIATING CONDITION MATRIX

RADIATION LEVELS

INITIATING CONDITION	EMERGENCY ACTION LEVEL	CLASS
RA3 Release of Radioactive Material or Increases in Radiation Levels that Impedes Operation of Systems Required to Maintain Safe Operation or to Establish or Maintain Cold Shutdown. **Modes: At all times**	Valid Readings on any of the following Area Monitors: **EAL-1** RT-8066 > 15 mR/hr (35' EAB) **EAL-2** RT-8058 > 5.00 E+3 mR/hr (10' MAB) RT-8060 > 5.00 E+3 mR/hr (10' MAB) RT-8061 > 5.00 E+3 mR/hr (10' MAB) RT-8062 > 5.00 E+3 mR/hr (10' MAB) RT-8063 > 5.00 E+3 mR/hr (29' MAB) RT-8077 > 5.00 E+3 mR/hr (60' MAB) RT-8084 > 5.00 E+3 mR/hr (-21' FHB) RT-8085 > 5.00 E+3 mR/hr (-21' FHB) RT-8086 > 5.00 E+3 mR/hr (-21' FHB) RT-8087 > 5.00 E+3 mR/hr (-21' FHB) RT-8090 > 5.00 E+3 mR/hr (68' FHB)	**ALERT**

RECOGNITION CATEGORY R
RADIOLOGICAL
INITIATING CONDITION MATRIX

RADIATION LEVELS

INITIATING CONDITION	EMERGENCY ACTION LEVEL	CLASS
RU2 Unexpected Increase in Plant Radiation Levels or Airborne Concentrations. **Modes: At all times**	**EAL-1** Valid Readings on any of the following Area Monitors greater than 1,000 Times 24 hr. average. RT-8052, (-11’ RCB) RT-8053, (-11’ RCB) RT-8054, (19’ RCB) RT-8055, (68’ RCB) RT-8056, (52’ RCB) RT-8057, (10’ EAB) RT-8058, (10’ MAB) RT-8059, (10’ MAB) RT-8060, (10’ MAB) RT-8061, (10’ MAB) RT-8062, (10’ MAB) RT-8063, (29’ MAB) RT-8064, (29’ MAB) RT-8065, (29’ MAB) RT-8066, (35’ EAB) RT-8067, (35’ EAB) RT-8068, (41’ MAB) RT-8069, (OSC) RT-8070, (41’ MAB) RT-8071, (41’ MAB) RT-8072, (41’ MAB) RT-8073, (41’ MAB) RT-8074, (41’ MAB) RT-8075, (41’ MAB) RT-8076, (60’ EAB) RT-8077, (60’ MAB) RT-8078, (60’ MAB) RT-8079, (60’ MAB) RT-8080, (41’ MAB) RT-8081, (68’ FHB) RT-8082, (60’ MAB) RT-8083, (41’ MAB) RT-8084, (-21’ FHB) RT-8085, (-21’ FHB) RT-8086, (-21’ FHB) RT-8087, (-21’ FHB) RT-8088, (30’ FHB) RT-8089, (68’ FHB) RT-8090, (68’ FHB) RT-8091, (68’ FHB) RT-8092, (29’ TGB) RT-8093, (29’ TGB) RT-8094, (TSC) RT-8096, (EOF) RT-8097, (68’ FHB) RT-8098, (60’ MAB) RT-8099, (60’ RCB) RT-8100, (35’ EAB) RT-8101, (35’ EAB) **EAL-2** Uncontrolled* loss of water level in the Spent Fuel Pool and Fuel Transfer Canal with all irradiated fuel assemblies remaining covered by water. **EAL-3** Uncontrolled* decrease of water level in the Refueling Cavity/ICSA with all irradiated fuel assemblies remaining covered with water. (Mode 6 Only).	**UE**

*Outside the immediate control of the operator

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

NOTE

- **IF any SG(s) are faulted OR INACTIVE THEN DO NOT transition to 0POP05-EO-ES03, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL.**
- **IF it is determined that a natural circulation cooldown and depressurization must be performed at a rate that may form a steam void in the vessel, THEN 0POP05-EO-ES03, NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL, should be used.**

___16 **INITIATE RCS Depressurization:**

ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
___a. CHECK CRDM vent fans - AT LEAST TWO RUNNING	a. PERFORM the following: 1) ESTABLISH RCS subcooling based on core exit T/Cs GREATER THAN 100°F per ADDENDUM 2, COOLDOWN CURVE WITH LESS THEN TWO CRDM VENT FANS RUNNING. 2) MAINTAIN upper head subcooling GREATER THAN 10°F per ADDENDUM 3, UPPER HEAD SUBCOOLING CURVE (Plant Computer display RC-12/8112).

Step 16 continued on next page.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

Step 16 continued from previous page.

3) IF upper head subcooling LESS THAN 10°F, THEN:

    a) OPEN reactor vessel head vent isolation valves.

    b) OPEN reactor vessel head vent throttle valve in one vent path.

    c) MAINTAIN pressurizer level BETWEEN 23% AND 33% by controlling charging flow.

    d) WHEN upper head subcooling GREATER THAN 20°F, THEN PERFORM the following:

        1) Close all reactor vessel head vent throttle valves.

        2) Close all reactor vessel head vent isolation valves.

4) GO TO Step 16.c.

___b. MAINTAIN RCS subcooling based on core exit T/Cs - GREATER THAN 85°F

___c. CHECK auxiliary spray in service

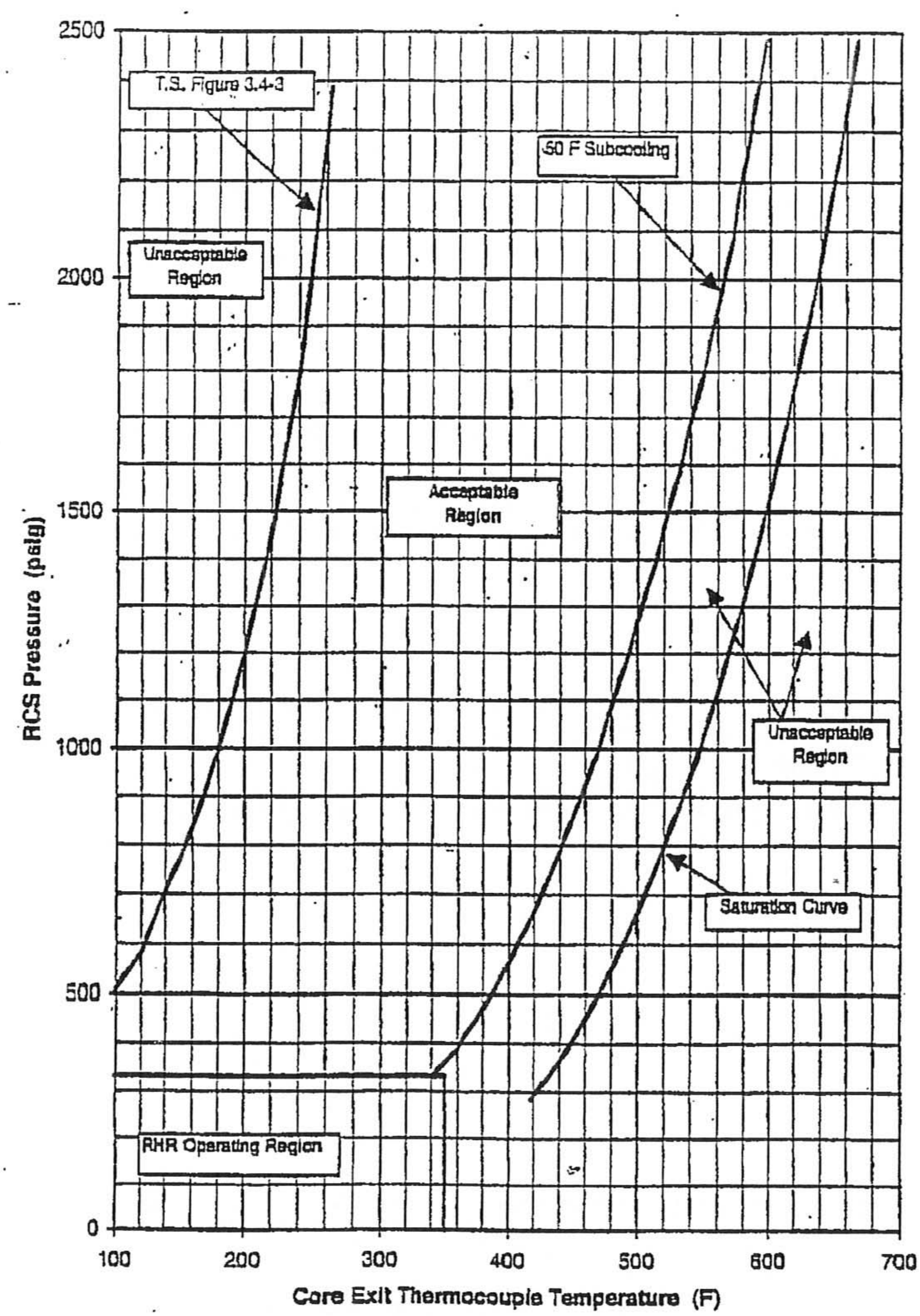
c. ENSURE normal spray valve - CLOSED

___d. USE auxiliary spray

d. USE one pressurizer PORV.

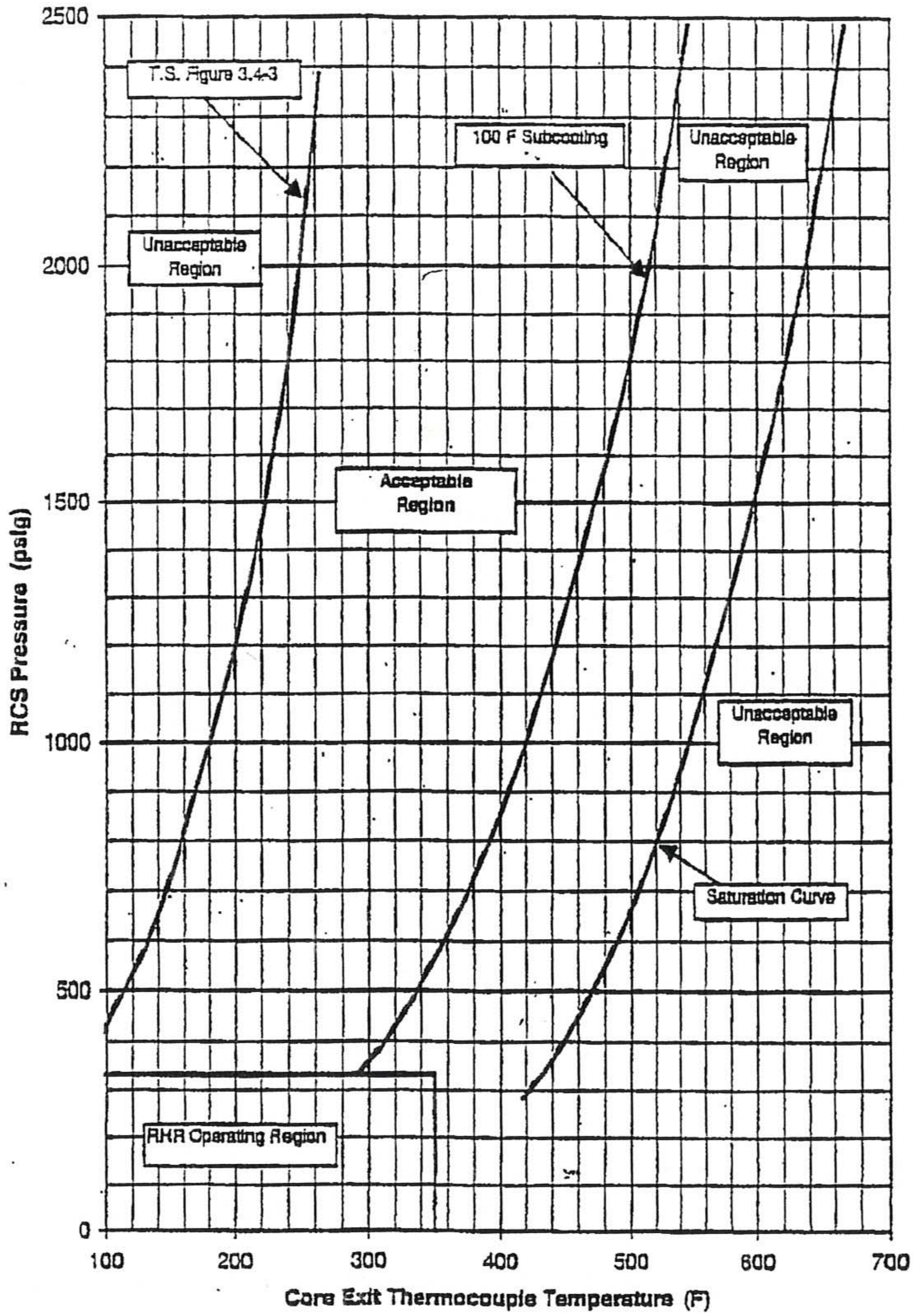
ADDENDUM 1
COOLDOWN CURVE WITH TWO OR MORE CRDM VENT FANS RUNNING

Cooldown Curve With Two Or More CRDM Vent Fans Runnings



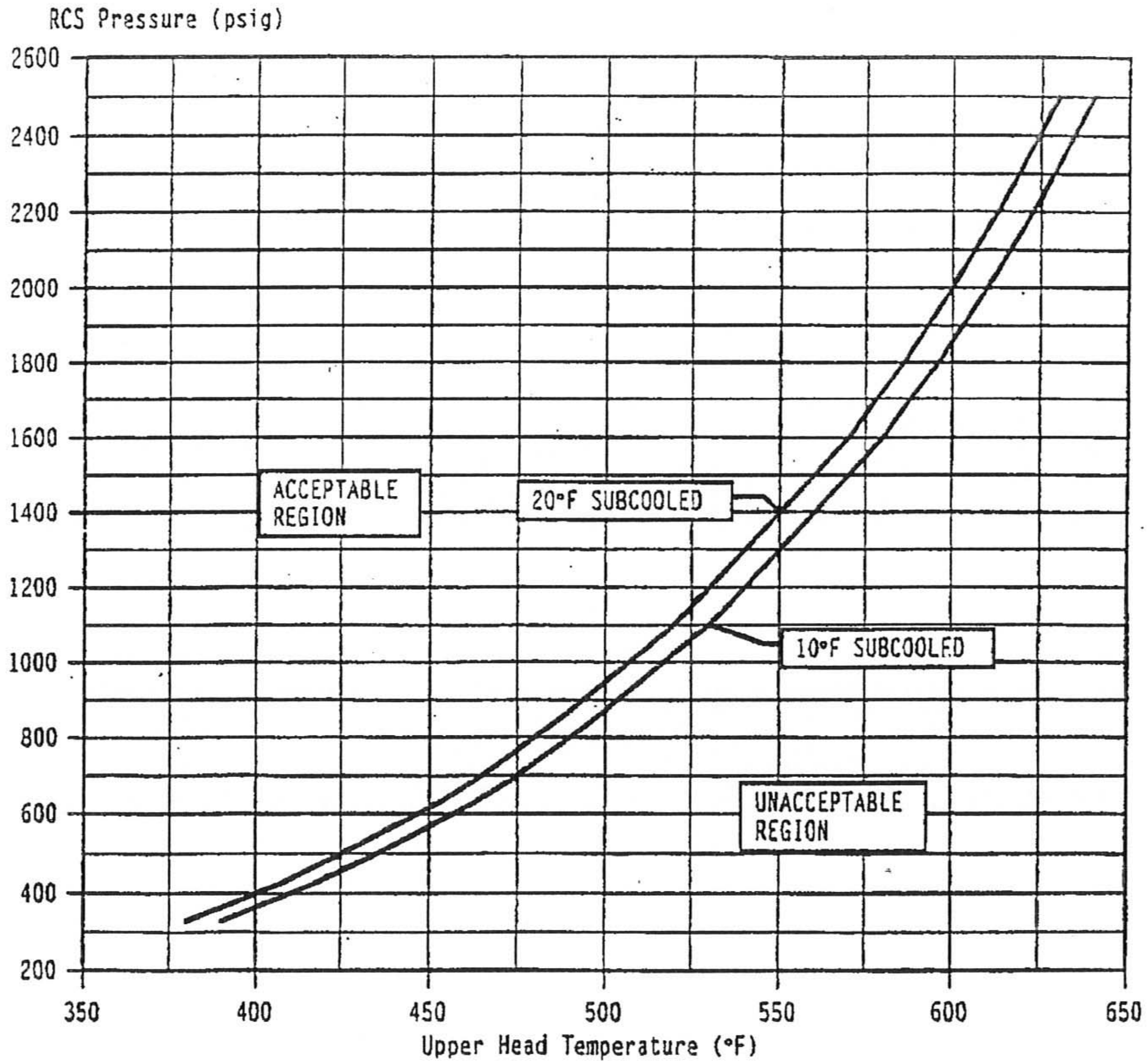
ADDENDUM 2
COOLDOWN CURVE WITH LESS THEN TWO CRDM VENT FANS RUNNING

Cooldown Curve With Less Than Two CRDM Vent Fans Running



ADDENDUM 3
UPPER HEAD SUBCOOLING CURVE

Upper Head Subcooling Curve



3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
13. Reactor Coolant Flow--Low					
a. Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6
14. Steam Generator Water Level--Low-Low	4/stm. gen.	2/stm. gen. in any operating stm. gen.	3/stm. gen. each operating stm. gen.	1, 2	6
15. Undervoltage--Reactor Coolant Pumps (Interlocked with P-7)	4-1/bus	2	3	1	6
16. Underfrequency--Reactor Coolant Pumps (Interlocked with P-7)	4-1/bus	2	3	1	6
17. Turbine Trip (Interlocked with P-9)					
a. Low Emergency Trip Fluid Pressure	3	2	2	1	6
b. Turbine Stop Valve Closure	4	2	3	1	6

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
18. Safety Injection Input from ESFAS	2	1	2	1, 2	9A
19. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2##	8
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	8
or					
P-13 Input	2	1	2	1	8
c. Power Range Neutron Flux, P-8	4	2	3	1	8
d. Power Range Neutron Flux, P-9	4	2	3	1	8
e. Power Range Neutron Flux, P-10	4	2	3	1, 2	8
f. Turbine Impulse Chamber Pressure, P-13	2	1	2	1	8
20. Reactor Trip Breakers	2	1	2	1, 2	9, 12
	2	1	2	3*, 4*, 5*	10

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
21. Automatic Trip and Interlock Logic	2	1	2	1, 2	9A
	2	1	2	3*, 4*, 5*	10

TABLE NOTATIONS

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

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

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

ACTION STATEMENTS

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

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

a. For Functional Units with installed bypass test capability,

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

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

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

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

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

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

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

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. Below the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above the P-6 Setpoint, and

b. Above the P-6 (Intermediate Range Neutron Flux Interlock) Setpoint but below 10% of RATED THERMAL POWER, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10% of RATED THERMAL POWER.

ACTION 4 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, suspend all operations involving positive reactivity changes. Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

ACTION 5 - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 72 hours, or immediately suspend all operations involving positive reactivity changes.

Note: Plant temperature changes or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

b. With the number of OPERABLE channels two less than the Minimum Channels OPERABLE requirement,

Immediately suspend all operations involving positive reactivity changes,

AND

Within 15 minutes isolate unborated water flow paths from the reactor makeup water system to the reactor coolant system,

AND

Perform either of the following:

Restore at least one channel to OPERABLE status within 1 hour,

OR

1. Within 2 hours secure each unborated water flow path to the reactor coolant system by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured,

   AND

2. Within 4 hours and once per 12 hours thereafter, verify SHUTDOWN MARGIN is within limits.

Note: Operations involving plant temperature changes may proceed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

ACTION STATEMENTS (Continued)

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. For Functional Units with installed bypass test capability, the inoperable channel may be placed in bypass, and must be placed in the tripped condition within 72 hours.

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

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

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

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

ACTION 7 - (Not Used)

ACTION 8 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

ACTION 9 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

ACTION 9A - a. With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, within 24 hours restore the inoperable channel to OPERABLE status, or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE.

b. With the number of OPERABLE channels more than one less than the Minimum Channels OPERABLE requirement, within 1 hour restore at least one inoperable channel to OPERABLE status or apply the requirements of the CRMP, or be in at least HOT STANDBY within the next 6 hours.

ACTION 10 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or open the Reactor Trip System breakers within the next hour.

ACTION 11 - (Not Used)

ACTION 12 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 9. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Specification 3.6.3;

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

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*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than at a frequency in accordance with the Surveillance Frequency Control Program.

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

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

ACTION:

> NOTE
>
> Entry and exit through the containment air lock doors is permitted to perform repairs on the affected air lock components.

a. With only one containment air lock door inoperable:

1. Verify the OPERABLE air lock door is closed within 1 hour and either restore the inoperable air lock door to OPERABLE status within 24 hours or lock the OPERABLE air lock door closed;

2. Operation may then continue provided that the OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and

b. With only the containment air lock interlock mechanism inoperable:

1. Verify an OPERABLE air lock door is closed within 1 hour and lock an OPERABLE air lock door closed within 24 hours;

2. Operation may then continue provided that an OPERABLE air lock door is verified to be locked closed at least once per 31 days;

3. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (Entry into and exit from containment is permissible under the control of a dedicated individual); and

c. With the containment air lock inoperable, except as specified in ACTION a. or ACTION b. above, immediately initiate action to evaluate overall containment leakage rate per Specification 3.6.1.2 and verify an air lock door is closed within 1 hour. Restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.

b. At a frequency in accordance with the Surveillance Frequency Control Program by verifying that only one door in each air lock can be opened at a time.

c. By verifying at a frequency in accordance with the Surveillance Frequency Control Program that the instrument air pressure in the header to the personnel airlock seals is $\geq$ 90 psig.

d. By verifying the door seal pneumatic system OPERABLE at a frequency in accordance with the Surveillance Frequency Control Program by conducting a seal pneumatic system leak test and verifying one of the following:

   1) That system pressure does not decay more than 1.5 psi from 90 psig minimum within 24 hours, or

   2) That system pressure does not decay more than .50 psi from 90 psig minimum within 8 hours.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two Source Range Neutron Flux Monitors* shall be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or operations that would cause introduction into the RCS of coolant with boron concentration less than required to meet the boron concentration of LCO 3.9.1.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at a frequency in accordance with the Surveillance Frequency Control Program,

b. A CHANNEL CALIBRATION, excluding the Neutron detectors, at a frequency in accordance with the Surveillance Frequency Control Program.

* An Extended Range Neutron Flux Monitor may be substituted for one of the Source Range Neutron Flux Monitors provided the OPERABLE Source Range Neutron Flux Monitor is capable of providing audible indication in the containment and control room.

SRO Exam Source

Bank

587	1147

Total: 2

New

1762	1763	1764	1765
1766	1767	1768	1769
1770	1771	1772	1774
1794	1845	1847	1848
1849	1850	1851	1852
1853	1854	1855	

Total: 23

Total Questions - 25

SRO Exam Answer Distribution

A

1848	1850	1854			

Total: 3

B

1147	1766	1772	1794	1845	1853

Total: 6

C

587	1762	1763	1767	1770	1847
1849	1851	1852			

Total: 9

D

1764	1765	1768	1769	1771	1774
1855					

Total: 7

Number of questions 25

LOT-17 NRC SRO Exam Difficulty

Bank#	1762	Difficulty	4	Bank#	1763	Difficulty	3	Bank#	1764	Difficulty	3
Bank#	1765	Difficulty	3	Bank#	1766	Difficulty	3	Bank#	1767	Difficulty	3
Bank#	1768	Difficulty	3	Bank#	1769	Difficulty	3	Bank#	1770	Difficulty	3
Bank#	1771	Difficulty	3	Bank#	1794	Difficulty	3	Bank#	1845	Difficulty	3
Bank#	1847	Difficulty	3	Bank#	1848	Difficulty	3	Bank#	1849	Difficulty	3
Bank#	1850	Difficulty	3	Bank#	1851	Difficulty	3	Bank#	1852	Difficulty	3
Bank#	1853	Difficulty	3	Bank#	1854	Difficulty	3	Bank#	1855	Difficulty	3
Bank#	587	Difficulty	3	Bank#	1147	Difficulty	3	Bank#	1772	Difficulty	3
Bank#	1774	Difficulty	3								

Average difficulty 3.04 **Number of questions** 25

LOT-17 NRC SRO Exam Question Level

F

587	1147	1768	1770
1771	1772	1850	1852

Total: 8

H

1762	1763	1764	1765
1766	1767	1769	1774
1794	1845	1847	1848
1849	1851	1853	1854
1855			

Total: 17

Total Questions - 25