 # 1 ☑ RO ☑ SRO Question ID: 5000001 Origin: New ☐ Memory Level

The plant was tripped from 100% power due to a report of a large steam leak in the East Penetration Area. Both MSIVs were manually closed immediately following the trip at the direction of the US. The Standard Post Trip Actions of EOP 2525 have been completed. The Shift Technical Advisor has just completed the first Safety Function Status Check for EOP 2526 and has determined that all safety functions are being satisfied.

Based on the following information, identify the procedure that must be implemented upon exit from EOP 2526.

- Pressurizer pressure	2010 psia, decreasing slowly
- Pressurizer Level	40%, decreasing slowly
- #1 SG Pressure	880 psia, decreasing slowly
- #2 SG Pressure	900 psia, steady
- SG levels	50% in both SG
- RCS Subcooling	97 degrees F, increasing slowly
- Rad monitors	indications normal, steady and not in alarm
- ADVs	closed
- Tcold	531 degrees F, slowly decreasing
- Thot	533 degrees F, slowly decreasing
- RCPs	operating
- Cntmt pressure	0.15 psig, steady

A OP-2207, Plant Cooldown ☐

B EOP-2532, Loss of Coolant Accident ☐

C EOP-2536, Excess Steam Demand Event ☑

D OP-2272C, Plant Operation in MODE 3 prior to Reactor Startup ☐

Justification

CHOICE (A) - NO
WRONG: Indications reveal an uncontrolled cooldown in progress as a result of the steam leak upstream of the MSIV. Exiting the EOPs during an uncontrolled cooldown is not an appropriate action. "EOP 2526 Reactor Trip Recovery Technical Guide", states that EOPs are exited and operating procedures are entered when the desired condition of the plant is determined, and a plant procedure exists to establish and maintain the plant in the desired condition. While a cooldown is desired, the specific guidance for conducting the cooldown during an excess steam demand event is contained in the optimal recovery guideline.
VALID DISTRACTOR: an applicant may think use of EOP-2536 is precluded unless the threshold of steam generator pressure less than 800 psia is met in EOP-2541, Appendix 1, "Diagnostic Flowchart".

CHOICE (B) - NO
WRONG: Indications are not consistent with a loss of coolant accident. Pressurizer pressure and level are decreasing. However, containment pressure is steady and radiation monitor indications are normal. SG pressure and RCS temperatures are decreasing, indicative of excessive heat removal.
VALID DISTRACTOR: an applicant could misinterpret the cause of lowering pressurizer pressure and level as a loss of coolant accident.

CHOICE (C) - YES
EOP-2541, Appendix 1, "Diagnostic Flowchart", directs use of the "appropriate optimal recovery guideline" following a single event diagnosis. The existence of an inaccessible steam leak is given in the question stem. The appropriate optimal guideline for a steam leak is EOP-2536, "Excess Steam Demand Event".

CHOICE (D) - NO
WRONG: Indications reveal an uncontrolled cooldown in progress as a result of the steam leak upstream of the MSIV. Exiting the EOPs during an uncontrolled cooldown is not an appropriate action. "EOP 2526 Reactor Trip Recovery Technical Guide", states that EOPs are exited and operating procedures are entered when the desired condition of the plant is determined, and a plant procedure exists to establish and maintain the plant in the desired condition. Current plant conditions show that the plant cannot be maintained in Hot Standby because an uncontrolled cooldown is in progress. The cooldown rate will increase as the decay heat generation rate drops.
VALID DISTRACTOR: an applicant may decide that, given all key parameters still within control bands specified in the EOP, that no further action is required other than to maintain the plant in a shutdown condition.

References

1. EOP 2526 Reactor Trip Recovery Technical Guide, Revision 15 (Pg 15 of 26)
2. EOP-2541, Appendix 1, "Diagnostic Flowchart", Revision 000 (10/2/03) (Pg 1 of 1)

NRC K/A System/E/A

System E02 Reactor Trip Recovery

Number EK1.2 **RO** 3.0 **SRO** 3.4 **CFR Link** (CFR: 41.8 / 41.10, 45.3)

Knowledge of the operational implications of the following concepts as they apply to the (Reactor Trip Recovery); Normal, abnormal and emergency operating procedures associated with (Reactor Trip Recovery).

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

2 ☑ RO ☑ SRO Question ID: 0054227 Origin: Bank ☐ Memory Level

The reactor has just tripped from 100% power. As PPO you are carrying out EOP-2525, Standard Post Trip Actions. You note the following conditions:

- Pressurizer pressure indicates 1800 psia and lowering
- Acoustic monitor indications are zero and steady
- Tavg indicates 535F and steady
- No rad monitor alarms are present
- Containment pressure is 0 psig and steady
- Pressurizer level is 33% and rising

What is the appropriate action to take?

A Isolate letdown. ☐

B Ensure MSI is actuated. ☐

C Stop RCPs as necessary. ☑

D Close PORV Block valves. ☐

Justification

CHOICE (A) - NO
WRONG: EOP-2525, "Standard Post Trip Actions", (step 3 contingency) directs the operator to manually operate charging and letdown if pressurizer level is not being restored to the 35-70% band. Level is at 33% and rising.
VALID DISTRACTOR: an applicant could decide to isolate letdown after incorrectly diagnosing the event as a pressurizer steam space leak.

CHOICE (B) - NO
WRONG: Indications are not consistent with an excess steam demand event. Pressurizer pressure is decreasing. However, average coolant temperature and containment pressure are steady and radiation monitor indications are normal.
VALID DISTRACTOR: an applicant could misinterpret the cause of lowering pressurizer pressure as an excess steam demand event.

CHOICE (C) - YES
With the given indications, there is no LOCA (no containment radiation monitor indications and no containment pressure rise), no ESD (steady Tavg), and no SGTR (no secondary radiation monitor indications). Therefore the primary pressure decrease must be due to a stuck open spray valve. EOP-2525, "Standard Post Trip Actions", directs stopping RCPs as necessary if RCS pressure cannot be restored and maintained between 2225 and 2300 psia and any spray valve will not close. "EOP-2525 Standard Post Trip Actions Technical Guide" states that differential pressure created by the RCPs provides the motive force for the pressurizer sprays and that securing the RCP will reduce the spray flow and the lowering of RCS pressure. Pressure control system lesson material (PLC-01-C) describes a stuck open spray valve event at MS2 on April 29, 1980. During this event operators stopped the 'A' RCP (Loop 1A) and then the 'B' RCP (Loop 1B). The uncontrolled depressurization was terminated after the 2nd reactor coolant pump was stopped.

CHOICE (D) - NO
WRONG: EOP 2525, "Standard Post Trip Actions", directs closing the PORV block valves if a PORV is open and pressure is less than 2250 psia. An open PORV at 1800 psia would be revealed by acoustic monitor indications. However, acoustic monitor indications are given as zero and steady.
VALID DISTRACTOR: an applicant may incorrectly attribute the event to an open PORV.

References

1. EOP-2525, "Standard Post Trip Actions", Revision 20 (2/22/01) (Pg 8, 9, 10 of 26)
2. EOP-2525 Standard Post Trip Actions Technical Guide, Revision 20 (Pg 11 of 38)
3. PLC-01-C, "Pressurizer Level & Pressure Control System" Lesson, Revision 3 (Pg 46, 47 of 61)

NRC K/A System/E/A

System 008 Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open)

Number AA2.02 **RO** 3.9 **SRO** 4.1 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: PZR spray valve position indicators and acoustic monitors

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

3 ☑ RO ☑ SRO Question ID: 5000002 Origin: Bank ☑ Memory Level

A principle difference between a Large Break LOCA (LBLOCA) and a Small Break LOCA (SBLOCA) is:

A Only the LBLOCA clears the RCP loop seal. ☐

B Only the LBLOCA causes core uncovery. ☐

C Only the SBLOCA requires heat removal from the S/Gs. ☑

D Only the SBLOCA results in peak clad temperatures > 1500F. ☐

Justification

CHOICE (A) - NO
WRONG: Plant-specific accident analysis shows both break categories can result in clearing the loop seal.
VALID DISTRACTOR: an applicant could reasonably assume that the smaller break, with a lower rate of mass loss, will not allow the loop seal to clear.

CHOICE (B) - NO
WRONG: Plant-specific accident analysis shows both break categories can result in core uncovery.
VALID DISTRACTOR: an applicant could reasonably assume that the smaller break, with a lower rate of mass loss, will not allow the loop seal to clear.

CHOICE (C) - YES
Plant-specific accident analysis shows both break categories can result in core uncovery. Further, "EOP 2532 Loss of Coolant Accident Technical Guide" (p1 of 18) states that for small breaks, heat removal via the flow out the break is not sufficient to provide cooling until at least the point where break uncovery occurs and, therefore, steam generator heat removal is required.

CHOICE (D) - NO
WRONG: Plant-specific accident analysis shows both break categories can result in peak cladding temperatures (PCT) in excess of 1500 degrees F. The limiting LBLOCA results in PCT of 1814F and the limiting SBLOCA results in PCT of 2061F.
VALID DISTRACTOR: an applicant could reasonably assume that the larger break, with a very rapid blowdown and subsequent rapid reflood will not provide sufficient time with core uncovered to allow PCT to exceed 1500F.

References

1. EOP-2532 Loss of Coolant Accident Technical Guide, Revision 21 (Pg 1 of 18)
2. Millstone Unit 2 UFSAR Section 14.6, "Decreases in Reactor Coolant Inventory", 2003 Revision (Pgs 14.6-14, -20, -21)
3. Source: INPO Bank - Q# 22448 - Used at Diablo Canyon 1, 10/1/2002

NRC K/A System/E/A

System 009 Small Break LOCA
Number EK2.03 **RO** 3.0 **SRO** 3.3* **CFR Link** (CFR 41.7 / 45.7)
Knowledge of the interrelations between the small break LOCA and the following: S/Gs

NRC K/A Generic

System
Number **RO** **SRO** **CFR Link**

 # 4 ☑ RO ☑ SRO Question ID: 5000003 Origin: New ☑ Memory Level

The reactor automatically tripped from full power. The US has just entered EOP-2525, "Standard Post Trip Actions". NO operator actions have been taken.

Using the attached copy of the SPDS display, identify the event that has occurred.

A Feed Line Break ☐

B Loss of Coolant Accident ☑

C Steam Generator Tube Rupture on #2 SG ☐

D Excess Steam Demand inside Containment ☐

Justification

CHOICE (A) - NO
WRONG: Event is LBLOCA.
VALID DISTRACTOR: Containment pressure is elevated.

CHOICE (B) - YES
SPDS display copied off of simulator 1 minute after initiating fail open of #2 FRV followed by LBLOCA concurrent with trip from full power

CHOICE (C) - NO
WRONG: Event is LBLOCA.
VALID DISTRACTOR: SG #2 level higher than SG #1 level.

CHOICE (D) - NO
WRONG: Event is LBLOCA.
VALID DISTRACTOR: Thot, Tcold much lower than normal post-trip

References

1. PPC-00-C, "Plant Process Computer System" Lesson, Revision 1 (12/22/03) (Pg 16, 17, 30 of 30)

NRC K/A System/E/A

System 011 Large Break LOCA

Number EA1.17 **RO** 3.5* **SRO** 4.1* **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and monitor the following as they apply to a Large Break LOCA: Safety parameter display system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

5 ☑ RO ☑ SRO Question ID: 5000004 Origin: New ☐ Memory Level

Given:
Plant has been operating at 55% power with normal parameters for the last 24 hours. Maintenance is being performed on the "A" Main Feedwater Pump. Multiple seal pressure alarms actuate. The following RCP seal pressure indications on C-04R are observed and do NOT appear to be changing over time:

- vapor seal pressure:
RCP 1A = 70 psig and fluctuating +- 5 psig
RCP 1B = 60 psig and fluctuating +- 5 psig
RCP 2A = 70 psig and fluctuating +- 5 psig
RCP 2B = 55 psig and fluctuating +- 5 psig

- upper seal pressure:
RCP 1A = 750 psig and fluctuating +- 30 psig
RCP 1B = 1600 psig and fluctuating +- 50 psig
RCP 2A = 1260 psig and fluctuating +- 40 psig
RCP 2B = 780 psig and fluctuating +- 40 psig

- middle seal pressure:
RCP 1A = 1460 psig and fluctuating +- 30 psig
RCP 1B = 2070 psig and fluctuating +- 80 psig
RCP 2A = 1440 psig and fluctuating +- 120 psig
RCP 2B = 1510 psig and fluctuating +- 30 psig

Identify the correct diagnosis and proper response from the choices below.

A "B" RCP lower and middle seals have failed and/or degraded. Trip reactor, then stop "B" RCP. ☐

B "B" RCP lower and middle seals have failed and/or degraded. Start controlled plant shutdown ☑

C "C" RCP lower and middle seals have failed and/or degraded. Trip reactor, then stop "C" RCP. ☐

D "C" RCP lower and middle seals have failed and/or degraded. Start controlled plant shutdown ☐

Justification

Indications given result in the following pump seal differential pressures:

"A" Pump (RCP 1A)
lower - 790
middle - 710
upper - 680

"B" Pump (RCP 1B)
lower - 180
middle - 470
upper - 1540

"C" Pump (RCP 2A)
lower - 810
middle - 180
upper - 1190

"D" Pump (RCP 2B)
lower - 740
middle - 730
upper - 725

Seal is considered failed if differential pressure is less than 200 psid and RCS between 2200 and 2300 psia. Seals are designed to operate with d/p less than 1500 psid indefinitely. With a seal failed if either of the two intact seals starts to pump (pressure oscillations greater than +- 300 psid), start a controlled plant shutdown. If one seal stage is failed and d/p across either of the two intact seal stages is changing at a slow rate (less than 10 psid every hour) then if any remaining seal stage d/p lowers to between 500 and 550 psid or rises to greater than 1500 psid, start a controlled shutdown. With change at a faster rate, d/p setpoint is higher.

CHOICE (A) - NO
WRONG: Trip action not required unless impending failure of all three seals.
VALID DISTRACTOR: an applicant may determine B RCP lower and middle seals failed and think correct action is to trip reactor.

CHOICE (B) - YES
Lower seal has failed (<200 psid). Middle seal meets degradation criteria (<500 psid). Procedure directs controlled shutdown.

CHOICE (C) - NO
WRONG: Lower seal does not meet failure or degradation criteria. Lower seal d/p is 810 psid. Procedure directs continued operation.
VALID DISTRACTOR: an applicant may determine C RCP lower and middle seals failed and think correct action is to trip reactor.

CHOICE (D) - NO
WRONG: Lower seal does not meet failure or degradation criteria. Lower seal d/p is 810 psid. Procedure directs continued operation.
VALID DISTRACTOR: an applicant may determine C RCP lower and middle seals failed.

References

1. OP-2301C, "Reactor Coolant Pump Operation", Revision 17 (11/6/03), Section 4.13 "RCP Seal Failure Determination" (Pg 34-38 of 45)

NRC K/A System/E/A

System 015/017 Reactor Coolant Pump (RCP) Malfunctions

Number AK2.10 **RO** 2.8* **SRO** 2.8 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: RCP indicators and controls

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

6 ☑ RO ☑ SRO Question ID: 5000005 Origin: New ☐ Memory Level

The plant is stable at 80% power with letdown flow control in manual. Charging and letdown flow are balanced.

An RCS leak occurs, resulting in a certain rate of pressurizer level decrease. The PPO stabilizes pressurizer level by ___________ the output of Letdown Flow Control HIC-110.

With NO makeup to the VCT, the rate of VCT level decrease will be approximately ______ times the rate at which the pressurizer level was decreasing.

A lowering, two ☑

B lowering, one-half ☐

C raising, three ☐

D raising, one-third ☐

Justification

Pressurizer volume per % indicated level => 66.44 gals/% (at 2250 psia)
VCT volume per % indicated level => 34 gals/%

CHOICE (A) - YES
Controller output must be lowered to reduce letdown flowrate. Rate of VCT level decrease will be 1.954 (or approximately two) times the prior rate of pressurizer level decrease.

CHOICE (B) - NO
WRONG: the pressurizer volume per % indicated level is almost twice that of the VCT.
VALID DISTRACTOR: applicant may think the rate of VCT level decrease will be 1/2 that of the pressurizer.

CHOICE (C) - NO
WRONG: the controller output must be lowered to reduce letdown flow and the pressurizer volume per % indicated level is almost twice that of the VCT.
VALID DISTRACTOR: applicant may think controller output must be raised.

CHOICE (D) - NO
WRONG: the controller output must be lowered to reduce letdown flow and the pressurizer volume per % indicated level is almost twice that of the VCT.
VALID DISTRACTOR: applicant may think controller output must be raised.

References

1. CVC-00-C, "Chemical and Volume Control System" Lesson, Revision 8/3 (Pg 24 of 165)
2. OP-2304C, "Make Up (Boration and Dilution) Portion of CVCS", Revision 21/9 (6/4/04), Section 4.6, "Batch Makeup to VCT" (Pg 23 of 78)
3. SP-2602A, "Reactor Coolant Leakage", Revision 5/7 (8/31/04), Attachment 1, "RCS Pressure vs. Pressurizer Volume" (Pg 16 of 20)

NRC K/A System/E/A

System 022

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.2 **RO** 4.0 **SRO** 3.5 **CFR Link** (CFR: 45.2)

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

7 ☑ RO ☑ SRO Question ID: 5000006 Origin: New ☐ Memory Level

The plant has begun a refueling outage and is currently in MODE 5. RCS is in reduced inventory and draining is in progress, with the following conditions:

- 'A' LPSI Pump => running
- SDC Total Flow (F306) => 1600 gpm
- RCS to SDC Temp (T351X) => 97 degrees F
- Time after shutdown => 120 hours
- No. 2 Hot Leg NR Lvl (L-122) => + 0.5 inches

As the draining continues the following indications are observed:
- LPSI PUMP A SUCTION PRESSURE LO annunciator lit (C-01, A-8)
- Oscillating 'A' LPSI Pump current

Identify which of the following accounts for these indications.

A 'A' LPSI Pump cavitation due to vortexing in Loop 2 Hot Leg ☑

B A' LPSI Pump is operating at shutoff head due to vortexing in Loop 2 Hot Leg ☐

C A' LPSI Pump cavitation due to fully open SI-306, SDC SYS TOTAL FLOW VALVE ☐

D 'A' LPSI Pump operating at shutoff head due to fully closed SI-306, SDC SYS TOTAL FLOW VALVE ☐

Justification

CHOICE (A) - YES
Vortexing causes air entrainment and leads to pump cavitation.

CHOICE (B) - NO
WRONG: Pump operating at shutoff head would indicate steady current and static head at pump suction.
VALID DISTRACTOR: an applicant may assume that vortexing will cause pump to operate at shutoff head conditions.

CHOICE (C) - NO
WRONG: SI-306 failing closed would tend to reduce flow rate, thereby reducing chance of cavitation. SI-306 failing open would have no effect on flowrate because the LPSI Loop Injection Valves were previously throttled to limit total flow to less than or equal to 1600 gpm. See Procedure OP-2310, (Pg 19-22 of 109)
VALID DISTRACTOR: an applicant may think that SI-306 failing open in this situation would result in a high flow condition.

CHOICE (D) - NO
WRONG: In MODE 5, at 120 hours after shutdown, decay heat load requires flow through SDC HX via SI-657. System flow may be reduced by failure closed of SI-306. However, flow would continue through the HX, ensuring pump does not run at shutoff head.
VALID DISTRACTOR: an applicant may assume that closure of SI-306 will force pump to run at shutoff head conditions.

References

1. SDC-00-C, "Shutdown Cooling System" Lesson, Revision 3/4 (Pg 47 of 79)
2. OP-2310C, "Shutdown Cooling System", Revision 22 (9/8/04), Section 4.4, "Reducing SDC Flow in Preparation for Reduced Inventory" (Pg 19-22 of 109)
3. OP-2301E, "Draining the RCS (IPTE), Revision 22 (11/4/03) (Pg 2 of 66)
4. AOP-2572, "Loss of Shutdown Cooling, Revision 9 (10/9/03) (Pg 4 of 67)

NRC K/A System/E/A

System 025 Loss of Residual Heat Removal System (RHRS)

Number AA2.07 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Pump cavitation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

8 ☑ RO ☑ SRO Question ID: 5000007 Origin: Bank ☐ Memory Level

The following conditions exist on Unit 2:

o The reactor is shutdown
o Both trains of SDC in service
o RCS temperature is 280°F
o RCS pressure is 160 psia
o RBCCW surge tank level is decreasing

What leak location will produce these indications?

A Letdown Heat Exchanger ☐

B Thermal Barrier Heat Exchanger ☐

C Shutdown Cooling Heat Exchanger ☐

D Reactor Building Component Cooling Water Heat Exchanger ☑

Justification

CHOICE (A) - NO
WRONG: Letdown pressure on letdown heat exchanger is greater than RBCCW pressure (160 psia vs 95 psig).
VALID DISTRACTOR: applicant may not understand that letdown pressure on letdown heat exchanger > RBCCW pressure.

CHOICE (B) - NO
WRONG: RCS pressure on thermal barrier heat exchangers is greater than RBCCW pressure on the same heat exchangers (160 psia vs 95 psig).
VALID DISTRACTOR: applicant may not understand that RCS pressure on thermal barrier > RBCCW pressure.

CHOICE (C) - NO
WRONG: With both trains of shutdown cooling in service, the SDC system pressure in the SDC HXs (~165 psig) exceeds that of RBCCW (~95 psig).
VALID DISTRACTOR: applicant may not understand that SDC pressure > RBCCW pressure.

CHOICE (D) - YES
Service water pressure on RBCCW heat exchanger is less than RBCCW pressure (~45 psig vs 95 psig). Maximum SW pump delta-P of 65 psid in surveillance procedure data sheet (SP-2612A)

References

1. RBC-00-C, "Reactor Building Closed Cooling Water System" Lesson, Revision 6 (Pg 44, 45 of 73)
2. SP-2612A-003, "Surveillance Form", Revision 1 (4/6/04) (Pg 3 of 7)
3. Source: INPO Bank - Q# 3392 - Used at Braidwood 1, 9/14/1998

NRC K/A System/E/A

System 026 Loss of Component Cooling Water (CCW)

Number AA1.05 **RO** 3.1 **SRO** 3.1 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: The CCWS surge tank, including level control and level alarms, and radiation alarm

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

9 ☑ RO ☑ SRO Question ID: 0053364 Origin: Bank ☐ Memory Level

Unit 2 is operating at 100% power, steady state. Pressurizer Pressure Transmitter PT-100Y is NOT selected for control and is out-of-service for repairs.

Pressure on the selected transmitter, PT-100X, begins rising as indicated on Pressurizer Pressure Controller PIC-100X and the Plant Process Computer. Both pressurizer spray valves begin to open. Pressure is decreasing on all Pressurizer Pressure Safety Channels.

Which one of the following actions, taken by themselves, would maintain pressure at approximately 2250 psia?

A Turn the pressurizer backup heater control switches to ON. ☐

B Turn the pressurizer proportional heater control switches to ON. ☐

C Place PIC-100X in MANUAL and lower its output as necessary. ☑

D Place PIC-100X in MANUAL and raise its output as necessary. ☐

Justification

CHOICE (A) - NO
WRONG: Turning all backup heaters on manually will not maintain pressure as the normal amount of spray valve flow available can override the output of all pressurizer heaters combined.
VALID DISTRACTOR: applicant may assume that the additional heaters will stop the pressure reduction.

CHOICE (B) - NO
WRONG: Turning proportional heaters on manually will not maintain pressure as the normal amount of spray valve flow available can override the output of all pressurizer heaters combined. Also, the proportional heaters cannot be placed in service by simply closing their breaker hand switches. Proportional heaters operate off of the pressure controller output signal.
VALID DISTRACTOR: applicant may assume that the additional heaters will stop the pressure reduction.

CHOICE (C) - YES
With the pressure transmitter failed/failing high, the controller must be set to manual and the output lowered far enough to cause the spray valves to close and the output of the proportional heaters to rise, thereby restoring pressure to normal.

CHOICE (D) - NO
WRONG: Raising the controller output will open the spray valves more, and cause pressure to drop faster.
VALID DISTRACTOR: applicant may think that raising controller output will increase heater output and reduce spray flow.

References

1. PLC-01-C, "Pressurizer Level and Pressure Control System" Lesson, Revision 3 (Pg 22 of 61)

NRC K/A System/E/A

System 027

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.7 **RO** 3.7 **SRO** 4.4 **CFR Link** (CFR: 43.5 / 45.12 / 45.13)

"Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation."

 # 10 ☑ RO ☑ SRO Question ID: 1000045 Origin: Bank ☑ Memory Level

A SGTR has occurred in #1 SG concurrent with a loss of off-site power. Initial cooldown on both RCS loops has been completed and #1 SG has been completely isolated.

What parameter and value would indicate that the RCS cooldown was too aggressive and that the loops had become uncoupled?

A #1 loop Tc greater than or equal to 5° F lower than #2 loop Tc. ☐

B #1 loop Th greater than or equal to 10° F higher than #2 loop Th. ☑

C #1 loop delta-P greater than or equal to 5 psi lower than #2 loop delta-P. ☐

D #1 SG pressure greater than or equal to 20 psi higher than #2 SG pressure. ☐

Justification

A: isolated SG pressure remains elevated as part of success strategy to minimize pri-to-sec leakage; C: natural circ delta-P is ~1/2 # or less in loop #2, can't get 5# less; D: once #1 SG is completely isolated there is no way for its Tc to be lower

CHOICE (A) - NO
WRONG: Once #1 SG is completely isolated, #1 loop Tc will remain higher.
VALID DISTRACTOR: applicant may think that the difference in loop temperatures is indicative of uncoupling.

CHOICE (B) - YES
Uncoupling of the two loops is indicated by failure of Th in the loop with the isolated steam generator to track Th in the operating loop. Hot leg temperatures differing by more than 10°F is an indication that the isolated steam generator is limiting RCS cooldown and depressurization. (2nd note in 1st note block, EOP-2534, Pg 24 of 64)

CHOICE (C) - NO
WRONG: Natural circ delta-P is ~1/2 # or less in loop #2, can't get 5# less.
VALID DISTRACTOR: applicant may assume that the additional heaters will stop the pressure reduction.

CHOICE (D) - NO
WRONG: Isolated SG pressure remains elevated as part of success strategy to minimize pri-to-sec leakage.
VALID DISTRACTOR: applicant may assume that the differences in SG pressure indicative of uncoupling.

References

1. EOP-2534, "Steam Generator Tube Rupture", Revision 22 (3/22/02) (Pg 24 of 64)

NRC K/A System/E/A

System 038 Steam Generator Tube Rupture (SGTR)

Number EK1.03 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR 41.8 / 41.10 / 45.3)

Knowledge of the operational implications of the following concepts as they apply to the SGTR: Natural circulation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

 # 11 ☑ RO ☑ SRO Question ID: 0071648 Origin: Bank ☐ Memory Level

The plant was operating normally at 100% power, when the following events occurred:

- Pressurizer Pressure, Level, and Reactor Coolant (RCS) Cold Leg Temperature (Tc) start dropping rapidly
- Reactor trips
- Main Steam Isolation (MSI) and Safety Injection Actuation Signal (SIAS) occur
- Reactor Coolant Pumps (RCP's) are secured
- Loop 2 Tc and Steam Generator (S/G) pressure are decreasing much faster than Loop 1 Tc and S/G pressure.
- Auxiliary Feedwater Actuation Signal (AFAS) has NOT actuated
- Containment pressure and temperature are increasing

Which of the following actions must be taken on Panel C-05 in accordance with EOP-2536, "Excess Steam Demand Event" to mitigate this event?

A Place #2 S/G Auxiliary Feedwater Isolation Air Assisted Check Valve Switch to CLOSE. ☐

B Shift #1 and #2 Auxiliary Feedwater Regulating Valve Controllers to MANUAL and CLOSED. ☐

C Place #1 and #2 SG Auto Permissive OVERRIDE/MAN/START RESET Switches to PULL-TO-LOCK. ☑

D Shift #2 S/G Auxiliary Feedwater Regulating Valve RESET/NORM/OVRD Switch momentarily to OVRD. ☐

Justification

CHOICE (A) - NO
WRONG: The air assisted check valves are designed to provide containment isolation in the event of an accident inside containment. These valves are are 6 inch swing checks that will prevent a reversal of flow. Normal AFW flow will open the valves.
VALID DISTRACTOR: an EOP Step (EOP-2536, Step 9.L, Pg 12 of 62) directs closing this valve in the event of a steam line break. Applicant may think that closing this valve will prevent AFW from reaching the SG.

CHOICE (B) - NO
WRONG: An auto actuation signal will open the AFW feed regulating valves even in the manual loading stations are in MANUAL and CLOSED.
VALID DISTRACTOR: applicant may assume that the valve will not automatically open when in MANUAL.

CHOICE (C) - YES
The AFW feed regulating valves will be closed until AFAS is actuated. Placing these switches in PULL-TO-LOCK prior to AFAS blocks the automatic initiation signal that opens the AFW feed regulating valves. (AFW-00-C, Pg 19 of 56)

CHOICE (D) - NO
WRONG: The RESET NORM OVRD switch will not prevent feeding the SG if Auto AFW trips after the RESET NORM OVRD was momentarily (spring return to normal) in OVRD.
VALID DISTRACTOR: applicant may think that once overriden, the valve will not react to an auto actuation signal until this same switch is taken to RESET.

References

1. AFW-00-C, "Auxiliary Feedwater System" Lesson, Revision 5 (Pg 19, 20 of 56)
2. EOP-2536, "Excess Steam Demand Event", Revision 20 (2/27/01) (Pg 12 of 62)

NRC K/A System/E/A

System E05 Excessive Heat Transfer

Number EA1.1 **RO** 3.9 **SRO** 4.2 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the (Excessive Heat Transfer) Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

12 ☑ RO ☑ SRO Question ID: 5000008 Origin: New ☐ Memory Level

The plant is operating at 100% power, steady state conditions when both MFW pumps trip. The reactor trips on low steam generator water level. The plant operates normally post-trip except that feed flow CANNOT be established through MFW or AFW systems.

19 minutes later the following conditions exist:

- pressurizer pressure => 2390 psia, increasing slowly
- RCS subcooling => 98 °F, decreasing slowly
- Thot => 584 °F, increasing slowly
- Tcold => 581 °F, increasing slowly
- #1 SG WR level => 35 inches, steady
- #2 SG WR level => 22 inches, steady
- MSIVs are closed
- All RCPs are stopped
- SIAS has been actuated
- 'A' and 'B' HPSI, 'A' and 'B' LPSI, 'A' and 'C' Charging Pumps are running
- Currently in EOP-2540D, "Functional Recovery of Heat Removal"

Which of the following is correct regarding required actions and consequences?

A Open PORVs, core uncovery is likely due to inadequate injection flow. ☑

B Open PORVs, core uncovery is NOT likely due to sufficient injection flow. ☐

C Depressurize RCS by steaming ADVs at max rate, core uncovery is likely due to inadequate injection flow. ☐

D Depressurize RCS by steaming ADVs at max rate, core uncovery is NOT likely due to sufficient injection flow. ☐

Justification

CHOICE (A) - YES
PORVs must be opened to initiate once-through cooling. Per TG-2540D, if the plant trips from power on low steam generator level following a loss of all feed, SG level could reach the once-through-cooling action point as early as 7 to 9 minutes after event initiation. In this case, 19 minutes have elapsed. The document also states that if once-through-cooling is not initiated before SGs are lost as a heat sink, core uncovery and possible core damage could result. Transition to once-through cooling is directed if SG WR level < 70 inches or RCS temperature rises > 5 °F. RCS temperature has risen to 584°F due to the delay in initiating once-through cooling. At this temperature, RCS pressure will stablize at ~1360 psia following the opening of the PORVs. HPSI tests show 1280 psig pump discharge pressure at minimum flow condition of 25 gpm. Given the conditions, no HPSI or LPSI flow will be injected. The 2 running charging pumps will inject 88 gpm, which is not sufficient to make up for inventory loss.

CHOICE (B) - NO
WRONG: Core uncovery is likely for the reasons given for the correct answer.
VALID DISTRACTOR: applicant may think that sufficient injection flow is available since both trains of ECCS pumps are running.

CHOICE (C) - NO
WRONG: Heat sink is no longer effective as indicated by elevated RCS temperatures.
VALID DISTRACTOR: applicant may not identify that heat sink is no longer effective.

CHOICE (D) - NO
WRONG: Heat sink is no longer effective as indicated by elevated RCS temperatures.
VALID DISTRACTOR: applicant may not identify that heat sink is no longer effective.

References

1. HPI-00-C, "High Pressure Safety Injection System" Lesson, Revision 6, (Pg 12 of 49)
2. CVC-00-C, "Chemical and Volume Control System" Lesson, Revision 8, (Pg 37 of 165)
3. EOP-2540D Functional Recovery of Heat Removal Technical Guide, Revision 18, (Pg 122 of 155)

NRC K/A System/E/A

System E06 Loss of Feedwater

Number EA1.2 **RO** 3.4 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the (Loss of Feedwater) Operating behavior characteristics of the facility.

NRC K/A Generic

System Number	RO	SRO	CFR Link

13 ☑ RO ☑ SRO Question ID: 0053332 Origin: Bank ☐ Memory Level

The plant tripped from 100% power. A Station Blackout has been diagnosed, and the appropriate EOP entered.

Which one of the following sets of conditions satisfy the requirements for stable natural circulation, two hours into the event?

A RCS Tcold => 445 degrees F and constant ☐
RCS Thot => 422 degrees F and going down
CET => 443 degrees F and going down
RCS Pressure => 1500 psia

B RCS Tcold => 395 degrees F and constant ☐
RCS Thot => 453 degrees F and going down
CET => 442 degrees F and going down
RCS Pressure => 1600 psia

C RCS Tcold => 458 degrees F and going down ☐
RCS Thot => 480 degrees F and going down
CET => 481 degrees F and going down
RCS Pressure => 700 psia

D RCS Tcold => 458 degrees F and constant ☑
RCS Thot => 470 degrees F and constant
CET => 469 degrees F and constant
RCS Pressure => 930 psia

Justification

CHOICE (A) - NO
WRONG: Choice (has Thot lower than Tcold. Natural circulation will not establish in reverse direction. Also, difference between Thot and CET is 11 degrees, which is in excess of 10 degree limit.
VALID DISTRACTOR: applicant may not recognize that Tcold must be lower than Thot.

CHOICE (B) - NO
WRONG: Loop delta-T is 58 degrees, which is in excess of 55 degree criteria. Also difference between Thot and CET is 11 degrees, which is in excess of 10 degree limit.
VALID DISTRACTOR: applicant may not know that maximum loop delta T is < 55 degrees.

CHOICE (C) - NO
WRONG: Subcooling is 25 degrees, which is below minimum subcooling criteria of 30 degrees.
VALID DISTRACTOR: applicant may not recognize that subcooling is less than the required minimum.

CHOICE (D) - YES
Thot is greater than Tcold. Loop delta-T is 12 degrees, all temperatures are decreasing. Subcooling is 65 degrees.

References

1. EOP-2528 "Loss of Offsite Power/Loss of Forced Circulation", Revision 15 (2/27/01), (Pg 8 of 36)
2. EOP-2541, Appendix 2, "Figures", Revision 1 (3/31/04), Pg (1 of 7)
3. Steam Tables - SUPPLY TO APPLICANT FOR USE DURING EXAM

NRC K/A System/E/A

System 055 Loss of Offsite and Onsite Power (Station Blackout)

Number EA2.02 **RO** 4.4 **SRO** 4.6 **CFR Link** (CFR 43.5 / 45.13)

Ability to determine or interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

14 ☑ RO ☑ SRO Question ID: 5000009 Origin: New ☑ Memory Level

Power level is at 92% when the SPO reports that the FEEDWATER REGULATING VALVE 2 LOCKED alarm (B-8, C05) is actuated and all four white indicating lights for #2 SG Feedwater Control are out.

Based on these indications, on direction of the US, the SPO will take the following action(s):

A Press RX TRIP TCBS pushbuttons and close #2 SG FRV Blocking Valve FW-42B to prevent overfill. ☐

B Press 'A' and 'B' SGFP MAN pushbuttons and control pump speed manually to maintain level in #2 SG. ☑

C Press LIC-5269, #2 SG FRV Controller MAN pushbutton and maintain #2 SG level within the desired operating band. ☐

D Press #2 SG FRV DOWNCOMER RESET pushbutton and control valve in manual to restore level to between 60 and 75%. ☐

Justification

Indications caused by a loss of Vital Instrument Bus VA-20. This bus supplies control power to #2 SG FRV. Loss of power will cause normally open solenoid valves to close on the air supply lines to the Main and Bypass FRVs, which will fail in "as-is" position. Each FRV has four normally lit white control status lights. They indicate low instrument air header pressure, low control air pressure, high or low controller output or loss of control power. All four lights will extinguish if power if Bus VA-20 is lost.

CHOICE (A) - NO
WRONG: The ARP directs the operator to maintain power level constant. Given the indications and the slow rate of power decrease, the operator will be able to control SG level by varying SGFP speed.
VALID DISTRACTOR: applicant may assume that the loss of control while conducting a downpower will require a manual reactor trip. If the reactor is tripped, action to isolate feedwater would be appropriate since the FRV will not close. The AOP for loss of the bus (2504D) contains a caution that warns operator the FRV will not close if a reactor trip occurs and states that since FRV fails 'as is', a SG level transient may occur. This caution prepares operator for one possible outcome, but the following procedural guidance makes it apparent that actions are available to control SG level, thereby avoiding the need for a reactor trip.

CHOICE (B) - YES
ARP directs operator, if necessary, to place both SGFPs in manual and to control level by pump speed. This action is necessary since without control power, the FRV cannot be remotely positioned.

CHOICE (C) - NO
WRONG: With a loss of control power, the FRV cannot be controlled remotely in auto or in manual.
VALID DISTRACTOR: the alarm response, written for multiple possible causes for a FRV lock condition, does direct manual control of the FRV. The applicant must correctly diagnose the cause of the problem based on given indications in order to determine the correct ARP actions.

CHOICE (D) - NO
WRONG: Downcomer reset will not affect FRV control until control power is restored.
VALID DISTRACTOR: the ARP, written for multiple possible causes for a FRV lock condition, does direct the operator to press the pushbutton to restore manual control. The applicant must correctly diagnose the cause of the problem based on given indications in order to determine the correct ARP actions.

References

1. ARP-2590D, Window B-8 (030), "FEEDWATER REGULATING VALVE 2 FAILURE", (2/12/04)
2. AOP 2504D, "Loss of 120 VAC Vital Instrument Panel VA-20" Revision 3 (6/24/04) (Pg 7,10,19 of 23)
3. OP-2385, "Feedwater Control System Operation", Revision 9 (2/7/02) (Pg 7 of 22 and Figure 5)

NRC K/A System/E/A

System 057

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.10 **RO** 3.0 **SRO** 3.1 **CFR Link** (CFR: 41.10 / 43.5 / 45.13)

Knowledge of annunciator response procedures.

15 ☑ RO ☑ SRO Question ID: 0055206 Origin: Bank ☑ Memory Level

A loss of 125 VDC Bus 201A causes a plant trip. Buses 25A and 24C fail to transfer to the RSST.

The 'A' D/G will ________________________.

A remain shutdown and CANNOT be started from the control room ☐

B receive an emergency start and immediately trip on overspeed ☐

C come up to speed on the electrical governor and automatically load Bus 24C ☐

D come up to speed on the mechanical governor and have only limited protective features available ☑

Justification

CHOICE (A) - NO
WRONG: The diesel will start and run on the mechanical governor.
VALID DISTRACTOR: Applicant may think diesel will remain shutdown.

CHOICE (B) - NO
WRONG: The diesel will start and run on the mechanical governor.
VALID DISTRACTOR: The only available protective feature on a loss of DC control power is mechanical overspeed. Applicant may think the overspeed trip will be challenged by the loss of DC control power.

CHOICE (C) - NO
WRONG: The diesel will start and run on the mechanical governor and will not automatically load bus.
VALID DISTRACTOR: Diesel is designed to auto start and auto load on a loss of power to Bus 24C. Applicant may think diesel will function as designed.

CHOICE (D) - YES
The diesel generator air start solenoid valves fail open on a loss of DC. The diesel will start and run on the mechanical governor with only the overspeed trip available; all other trips need DC to operate. The diesel output breaker will not close without DC control power, so the diesel can not provide power to the bus. Question requires applicant to understand effects of loss of DC on the EDG and on Bus 24C.

References

1. AOP 2505A, "Loss of Vital 125 VDC Bus 201A" Revision 1 (2/12/03) (Pg 27, 28 of 47)
2. LVD-00-C, "125 VDC/120 VAC" Lesson, Revision 5 (Pg 52 of 77)

NRC K/A System/E/A

System 058 Loss of DC Power

Number AK3.01 **RO** 3.4* **SRO** 3.7 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.1)

Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: Use of dc control power by D/Gs

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

16 ☑ RO ☑ SRO Question ID: 5000010 Origin: New ☐ Memory Level

Plant is operating in MODE 3. 'C' Service Water Pump is out of service for motor maintenance. 'B' Service Water Pump is supplying the 'B' Service Water header. Bus 24E is aligned to Bus 24D. 'B' RBCCW Heat Exchanger is aligned to provide minimum flow for the 'A' Service Water header. Long Island Sound water temperature is 37°F. EDG SW Bypass Valves SW-231A and SW-231B are being maintained closed because of an issue with the adequacy of valve actuator spring pressure.

125 VDC Panel DV10 de-energizes due to a fault. You now have the following indications:

PPC Points:
F6433 'A' RBCCW HX SW Flow => 8885 gpm
F6434 'B' RBCCW HX SW Flow => 795 gpm
F6435 'C' RBCCW HX SW Flow => 8965 gpm

Local EDG SW Flows:
FIC-6397 'A' EDG => 850 gpm
FIC-6389 'B' EDG => 100 gpm

Assuming NO operator actions related to Service Water, which of the following is correct concerning these indications?

A 'B' EDG SW flow is LOWER than normal because SW-89B, DG TCV is closed. ☐

B 'A' EDG SW flow is HIGHER than normal because of a rupture downstream of 'A' EDG SW flow transmitter. ☐

C 'A' RBCCW HX SW flow is LOWER than normal because of diversion of flow to the in-service TBCCW HX. ☐

D 'C' RBCCW HX SW flow is HIGHER than normal because of a rupture downstream of 'C' RBCCW HX SW flow transmitter. ☑

Justification

Pre-event plant configuration:
Both EDGs shutdown, SW flow = ~150 gpm per EDG
Fac 1 aligned to 'A' RBCCW (400 gpm), 'A' TBCCW (1700 gpm) and 'B' TBCCW (2050 gpm for minimum flow purposes ,man vlv throttled)
Fac 2 aligned to 'C' RBCCW (400 gpm), 'B' RBCCW (1150 gpm for minimum flow purposes, man vlv SW-9B throttled), and 'C' TBCCW (1700 gpm)
Total SW flow = ~3000 to 4000 gpm per header
'B' and 'C' SW pumps both powered from Facility 2 PS (24D/24E).

Expected plant response to the loss of DC Panel DV10:
- MSIVs close, plant trips
- letdown isolates
- Facility 1 'A' EDG starts, does not load
- TCBs 1 and 3 open
- Facility 1 RBCCW flow balance is disrupted due to numerous valves failing
  - 'A' RBCCW SW Outlet TCV, SW-8.1A (TV-6308) fails open
  - EDG TCV, SW-89A (HV-6389) fails open
  - EDG Bypass Valve, SW-231A (FY-6341) fails closed
- Bus 24C loses control power, fails to transfer to RSST, de-energizes

Discussed system conditions with Dan Pantalone of MS2 on 11/19/04. Typical system flow conditions are expected operator knowledge. Plant SW conditions at power for winter operation (32-37°F Sound temp):
- currently operate with EDG Bypass Vlvs closed even when EDG is shutdown, based on issue with valve actuator spring force, when bypass was used, flow was 3000 gpm
- EDG SW flow with engine shutdown = 150 gpm, EDG flow with engine operating 1200 gpm
- RBCCW SW flow for in-service HX = 300 to 500 gpm, based on relatively small heat load at power and cold heat sink
- RBCCW SW flow for standby HX = 1000 to 1300 gpm, based on maintaining adequate minimum system flow, man vlv throttled
- TBCCW SW flow for in-service HX = 1500 to 2000 gpm, based on relatively large TB heat load at power
- TBCCW SW flow for standby HX = 2000 gpm, based on maintaining adequate minimum system flow, man vlv throttled
- Total SW Header flow = 3000 to 4000 gpm

CHOICE (A) - NO
WRONG: Loss of DV10 causes a reactor trip. However, Facility 2 equipment should not be challenged by loss of DV10 or the subsequent trip. Valve SW-89B was closed before the event and will remain closed throughout the event. Flow may be slightly reduced because of diverted flow to the RBCCW HX leak. However, flow given is still approximately normal for the EDG in standby due to typical leakage flow through the valve.

VALID DISTRACTOR: Applicant may think flow is reduced because TCV is closed.

CHOICE (B) - NO
WRONG: The DC panel failure results in 'A' EDG TCV failing open. DG Bypass is being maintained closed and fails closed. Flow will be higher than before the event because of open TCV. However, flow given is almost as expected during EDG operation, only somewhat reduced because of diverted flow to open RBCCW HX TCV. No leak is indicated.
VALID DISTRACTOR: Applicant may think the given flow is higher than normal due to a leak.

CHOICE (C) - NO
WRONG: Winter mode is selected to rout HX discharge through 6 inch bypass line. However, the loss of DC Panel DV10 causes the outlet valve (TV-6308) in the 14 inch line to fail open. Flow will be higher than prior to the event due to the valve opening.
VALID DISTRACTOR: Applicant may think flow is lower than prior to the event due to increased flow through the TBCCW heat exchanger. TBCCW TCVs do not fail on loss of DV-10.

CHOICE (D) - YES
Flow is normally reduced during winter conditions. All HX flow should be routed through the 6 inch outlet bypass line. Indicated flow is significantly higher than would be expected with the Sound at 37°F.

References

1. SWS-00-C, "Service Water System" Lesson, Revision 5 (Pg 16, 49 of 59)
2. LVD-00-C, "125 VDC/120 VAC" Lesson, Revision 5 (Pg 52, 53 of 77)
3. AOP-2505A, "Loss of Vital 125 VDC Bus 201A" Revision 1 (2/12/03) (Pg 20, 22 of 47)

NRC K/A System/E/A

System 062 Loss of Nuclear Service Water

Number AA2.01 **RO** 2.9 **SRO** 3.5 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: Location of a leak in the SWS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

17 ☑ RO ☑ SRO Question ID: 2100000 Origin: Bank ☐ Memory Level

A Xenon free, End-Of-Life (EOL) reactor startup is in progress.
RCS temperature is being maintained on the 'A' Steam Bypass valve. NO dilution is in progress.
Critical data is recorded and CEAs have been manually withdrawn 5 steps to raise reactor power to the POAH.
The SPO notes all 'High Power Trip Resets' on C04 are lit.

Which of the following actions is required for the given plant condition?

A Start an additional AFW pump to maintain SG levels. ☐

B Insert CEAs to their 10E-4 position to null out the positive dpm startup rate. ☐

C Operate Rx Trip pushbuttons due to CEAs stuck in continuous withdrawal. ☑

D Depress RPS High Power Trip Resets and open MSIV bypasses to restore Tave to 532°F. ☐

Justification

CHOICE (A) - NO
WRONG: Reactor trip is required for an uncontrolled rod withdrawal. Existing AFW alignment will provide sufficient flow post-trip to maintain SG levels.
VALID DISTRACTOR: Applicant may mistakenly think that AFW flow should be increased to match steam demand.

CHOICE (B) - NO
WRONG: Power is abnormally high, indicative of an large uncontrolled positive reactivity addition. A reactor trip is required.
VALID DISTRACTOR: Applicant may inappropriately apply operating procedure guidance for actions upon reaching POAH.

CHOICE (C) - YES
Hi-Power resets first light at ~9% power, much too high for a 1/2 DPM SUR to achieve under the stated conditions. All other sources of positive reactivity are ruled out by stated conditions. A reactor trip is required. From OP-2202, "If at anytime during this startup the condition of the reactor and its responses are not understood, and controlled by the operators, the reactor must immediately be tripped and the actions of EOP-2525, Standard Post Trip Actions performed."

CHOICE (D) - NO
WRONG: Power is abnormally high, indicative of an uncontrolled positive reactivity addition. A reactor trip is required.
VALID DISTRACTOR: Applicant may inappropriately apply operating procedure guidance for normal power escalation.

References

1. OP-2202, "Reactor Startup IPTE" Revision 20 (5/5/04) (Pg 2 of 47)

NRC K/A System/E/A

System 001 Continuous Rod Withdrawal

Number AA1.05 **RO** 4.3 **SRO** 4.2 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal: Reactor trip switches

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

 # 18 ☑ RO ☑ SRO Question ID: 5000072 Origin: New ☐ Memory Level

Unit 2 is operating at 100% power when the following alarms are received:

CEA MOTION INHIBIT
CEA REG GRPS WP
CEA REG GRP WP BKUP

There are NO changes in indicated position for any CEAs except for Shutdown Bank 'B' CEA #6, which indicates the following:
- red, amber and blue mimic lights out
- green mimic light lit
- PPC shows position at 180 steps
- CEAPDS shows position at 0 steps

Identify the position of CEA #6 and the reason for the conflicting indications.

A CEA #6 is fully inserted. A CEA drop reed switch has failed. ☑

B CEA #6 is fully withdrawn. A CEA drop reed switch has failed. ☐

C CEA #6 is fully inserted. A CEA reed switch position transducer switch has failed. ☐

D CEA #6 is fully withdrawn. A CEA reed switch position transducer switch has failed. ☐

Justification

CHOICE (A) - YES
Each of the 61 CEAs has 4 stacks of reed switches. One group is a stack of 97 switches connected to the reed switch position tranducer. This device generates a signal sent to the CEAPDS to display CEA position on a CRT on C-04. A second group consists of a reed switch which actuates when the CEA is fully inserted. This switch activates the dropped CEA annunciator, lights that CEA's core mimic amber light, and rezeros the PPC indication for that CEA at zero steps. A third group consists of the switch responsible for the Upper Electrical Limit signal and turns on the red light when CEA is > 180 steps. The fourth group consists of the switch responsible for the Lower Electrical Limit signal and turns on the green light when the CEA is < 1 step. CEA #6 is fully inserted. Since the drop reed switch has failed, there is no amber drop light. The PPC does not get a signal to reset position indication to 0

CHOICE (B) - NO
WRONG: CEA #6 is dropped.
VALID DISTRACTOR: May think the CEA is fully withdrawn based on lack of deviation alarms and PPC indication at 180 steps.

CHOICE (C) - NO
WRONG: CEA reed switch position transducer has not failed. Position indication on CEAPDS is correct.
VALID DISTRACTOR: May think CEA reed switch position transducer provides the PPC reset to zero signal.

CHOICE (D) - NO
WRONG: CEA is fully inserted.
VALID DISTRACTOR: May think the CEA reed switch position transducer is responsible for the CEA bottom light.

References

1. CED-01-C Rev 4 pages 30-31

NRC K/A System/E/A

System 003 Dropped Control Rod

Number AA2.01 **RO** 3.7 **SRO** 3.9 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod position indication to actual rod position

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

19 ☑ RO ☑ SRO Question ID: 0054389 Origin: Bank ☑ Memory Level

The plant is in normal operation at 100% power, when a Fire System Trouble annunciator is received on C06/7. Panel C-26 is checked and a PEO is subsequently dispatched to the West DC Switchgear Room.

The PEO reports the following:
- Two Ion Chamber smoke detectors are in alarm.
- The Halon strobe lights and horn are pulsating slowly.

Based on these conditions, what is the status of the West DC Switchgear Room Halon System?

A It is alarming as a warning of a potential discharge if a photoelectric detector is activated. ☑

B It is presently discharging or completed discharging to the West DC Switchgear Room. ☐

C It is warning that a discharge to the West DC Switchgear Room will occur after timer countdown. ☐

D It is in an alarmed state, should have already discharged, but a system malfunction has occurred. ☐

Justification

Existing bank comments: OP 2341A (Rev. 13); Discussion section and ARP 2590I, for Zone 45.$$$$[Copied from Item No '4990' on 10/28/97 By RLC]

CHOICE (A) - YES
The East and West DC switchgear rooms require two zones (one photoelectric smoke detector and one ion smoke detector) to initiate a halon release. Activation of one smoke detector, ion or photoelectric, will cause the strobe and horn to pulse slowly.

CHOICE (B) - NO
WRONG: The East and West DC switchgear rooms also require two zones (one photoelectric smoke detector and one ion smoke detector) to initiate a halon release. Activation of one smoke detector, ion or photoelectric, will cause the strobe and horn to pulse slowly.
VALID DISTRACTOR: an applicant may think that the pulsating horn announces a discharge in progress.

CHOICE (C) - NO
WRONG: Activation of a second smoke detector of the opposite type, but in the same room, will cause the strobe and horns for the affected room to pulse QUICKLY. The flashing lights will operate, and a 60 second pre-discharge time delay will begin. Upon expiration of the time delay the Halon System will discharge and the strobe and horn will sound steadily.
VALID DISTRACTOR: an applicant may think that the SLOWLY pulsating horn and strobe light warn of a timer countdown to discharge halon.

CHOICE (D) - NO
WRONG: Activation of a second smoke detector of the opposite type, but in the same room, will cause the strobe and horns for the affected room to pulse quickly. The flashing lights will operate, and a 60 second pre-discharge time delay will begin. Upon expiration of the time delay the Halon System will discharge and the strobe and horn will sound steadily.
VALID DISTRACTOR: an applicant may think that the pulsating horn and strobe lights indicate that a system malfunction has occurred.

References

1. OP 2341A, "Fire Protection System", Revision 15 (2/26/04) (Pg 3 of 61)
2. ARP 2590I, "Alarm Response for Fire Panel, C-26" (Zone 45), Revision 02 (9/9/04) (Pg 67-70, 97 of 106)

NRC K/A System/E/A

System 067 Plant fire on site

Number AA1.09 **RO** 3.0 **SRO** 3.3 **CFR Link** (CFR 41.7 / 45.5 / 45.6)

Ability to operate and / or monitor the following as they apply to the Plant Fire on Site: Plant fire zone panel (including detector location)

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

20 ☑ RO ☑ SRO Question ID: 5000012 Origin: New ☑ Memory Level

The plant is operating at 55% power for maintenance on a MFW pump.

A fire breaks out in Control Room Panel C-01. Smoke is filling the control room. The Shift Manager orders a control room evacuation.

What action is required to be performed from the control room prior to evacuation if time permits?

A Trip the reactor at C-04. ☑

B Open gravity feed valves at C-02. ☐

C Transfer in-house buses to the RSST at C-08. ☐

D Place both heater drain pumps in PULL-TO-LOCK at C-05. ☐

Justification

CHOICE (A) - YES
Fire Procedure AOP-2579A directs a reactor trip from C-04 (Pg 6 of 64)

CHOICE (B) - NO
WRONG: because AOP-2579A does not direct this action from the control room. The procedure directs lining up gravity feed from the BASTs locally (Step 21, Pg 15 of 64).
VALID DISTRACTOR: because the procedure directs opening these valves locally. The applicant may think this is an action to be performed remotely prior to exiting the control room.

CHOICE (C) - NO
WRONG: because AOP-2579A does not direct this action from the control room. Per the procedure the operator will contact CONVEX and have them de-energize RSST 15G-22S (Step 15, Pg 10 of 64)
VALID DISTRACTOR: because the non-fire shutdown from outside the control room procedure (AOP-2551) directs the transfer of in-house buses to the RSST prior to evacuation (Step 3.2, Pg 5 of 21).

CHOICE (D) - NO
WRONG: because AOP-2579A does not direct this action from the control room.
VALID DISTRACTOR: because the non-fire shutdown from outside the control room procedure (AOP-2551) directs the transfer of in-house buses to the RSST prior to evacuation (Step 3.2, Pg 5 of 21).

References

1. AOP-2579A, "Fire Procedure for Hot Standby Appendix R Fire Area R-1", Revision 9 (9/15/04) (Pg 6, 10, 15 of 64)
2. AOP-2551, "Shutdown from Outside the Control Room", Revision 9 (1/16/03) (Pg 5, 6 of 21)

NRC K/A System/E/A

System 068 Control Room Evacuation

Number AK2.02 **RO** 3.7 **SRO** 3.9 **CFR Link** (CFR 41.7 / 45.7)

Knowledge of the interrelations between the Control Room Evacuation and the following: Reactor trip system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

21 ☑ RO ☑ SRO Question ID: 5000014 Origin: New ☐ Memory Level

Unit 2 is at 64% power and increasing power to 100% at 3% per hour following a refueling outage.

While working at Panel C-03, an I&C technician stumbles and accidently trips 'A' and 'B' RCPs.

Assuming NO operator action and all equipment responds as designed, compare SG1 with SG2 approximately 3 minutes after the pumps are stopped. For SG2, which of the following sets of conditions describes the expected conditions for the parameters listed below?

Steam Pressure	Steam Flow	FRV Bypass Position

A HIGHER, HIGHER, FURTHER OPEN ☐

B HIGHER, HIGHER, EQUAL ☑

C EQUAL, LOWER, FURTHER CLOSED ☐

D EQUAL, EQUAL, EQUAL ☐

Justification

CHOICE (A) - NO
WRONG: Feed flow response for both steam generators is the same. The response is driven by the turbine trip signal, not SG level.
VALID DISTRACTOR: because the applicant may think that feed flow will be higher to maintain SG level with a higher steaming rate.

CHOICE (B) - YES
The reactor protection system will initiate an RCP underspeed trip for 2 pumps at <830 rpm and an automatic reactor trip on low reactor coolant flow in Loops 1A and 1B at a 92% flow setpoint. Three minutes after the event the post-trip decay heat is being removed via turbine bypass valves. However, only Loop 2 will be steaming. Flow in Loop 1 will be reversed, with Tcold pumped backward through Loops 1A and 1B. The higher rate of heat transfer from Loop 2 will result in HIGHER STEAM FLOW from SG2. Feed flow post-trip is determined by the design response of the feedwater control system. The FRVs ramp closed in manual at programmed rate on turbine trip signal. Bypass FRVs ramp open automatically to 40% over 3 minutes. Althought SG levels may be different, both SGs will have the EQUAL FEED FLOW rate. Steam PRESSURE will be HIGHER in SG2 because of the additional energy transfer. Pressure in SG 1 will be at saturation pressure for Tcold (900 psia at Tc=532°F)

CHOICE (C) - NO
WRONG: SG2 pressure will be higher because of the Thot inlet water, vs Tcold inlet water on SG1.
VALID DISTRACTOR: because the applicant may think pressures will be equal due to the relatively low rate of heat transfer post-trip.

CHOICE (D) - NO
WRONG: SG pressure will be higher because of the greater energy transfer rate.
VALID DISTRACTOR: because the applicant may think pressures will be equal due to the relatively low rate of heat transfer post-trip.

References

1. FWC-01-C, "Feedwater Control System", Revision 2 (3/22/04) (Pg 22, 23 of 46)
2. RPS-01-C, "Reactor Protection System", Revision 6 (9/15/00) (Pg 20 of 80)
3. Main Steam Print 25203-26002-1 (swing check valves downstream of MSIVs)

NRC K/A System/E/A

System 003 Reactor Coolant Pump System (RCPS)

Number K5.04 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implications of the following concepts as they apply to the RCPS: Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

22 ☑ RO ☑ SRO Question ID: 5000015 Origin: New ☑ Memory Level

The reactor is shutdown, maintaining Hot Standby conditions.

The following are indications for RCP 1A:
- Motor Vibration ==> 0.001 inches peak to peak
- Seal Bleed-Off Flow ==> 0.95 gpm
- Upper Oil Reservoir Level ==> 82.5%
- Motor Stator Temperature ==> 265°F

Given these conditions, identify the correct response related to conditions of RCP 1A.

A Trip pump because motor vibration exceeds limit ☐

B Trip pump not required because nothing exceeds limit ☐

C Trip pump because oil reservoir level exceeds limit ☐

D Trip pump because stator temperature exceeds limit ☑

Justification

CHOICE (A) - NO
WRONG: Motor vibration alarm setpoint is 0.002 to 0.005 inches peak-to-peak. In this range an other groups are contacted for determination as to whether or not pump should be shutdown.
VALID DISTRACTOR: because the applicant may think that vibration is excessive and requiring a pump shutdown.

CHOICE (B) - NO
WRONG: Trip is required due to high stator temperature.
VALID DISTRACTOR: because the applicant may all parameters within specification

CHOICE (C) - NO
WRONG: Oil level is in expected range (75 to 85%). The high level alarm actuates at 87.5%.
VALID DISTRACTOR: because the applicant may think oil level is excessive.

CHOICE (D) - YES
RCP stator temperature is normally 160 to 180 degrees F. The high motor stator temperature alarm actuates at 260 degrees F. Per the alarm response procedure, operators are directed to trip the plant and then the pump above 260 degrees F.

References

1. ARP-2590B-067, "RCP A VIBRATION HI ", Revision 0
2. ARP-2590B-066, "RCP A STR TEMP HI", Revision 0
3. ARP-2590B-082, "RCP A UPR OIL RSVR LEVEL HI", Revision 0
4. CVC-00-C, "Chemical and Volume Control System" Lesson, Revision 8/3 (Pg 14 of 165)

NRC K/A System/E/A

System 003 Reactor Coolant Pump System (RCPS)

Number A1.03 **RO** 2.6 **SRO** 2.6 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RCPS controls including: RCP motor stator winding temperatures

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

23 ☑ RO ☑ SRO Question ID: 0055022 Origin: Bank ☐ Memory Level

You have just taken the shift as the PPO with the following plant conditions:

- Plant is shutdown and in Mode 4
- RCS temperature is 290 degrees
- RCS pressure is 300 psia
- SDC warmup was initiated about ten minutes prior to your taking the shift

The Aux. Bldg. PEO calls up and reports that the "B" LPSI pump is making abnormal noises and the pump casing feels hot to the touch. He also reports that the "A" LPSI pump has normal running indications.

On C-01 the "B" LPSI pump motor current is lower than normal, and lower than the amps on the "A" LPSI pump.

The probable cause of this condition is that "B" LPSI Pump ___________________

A has a failing pump bearing. ☐

B is operating at runout conditions. ☐

C has a seated discharge check valve. ☑

D is experiencing a vortex in the SDC suction line. ☐

Justification

CHOICE (A) - NO
WRONG: A failling bearing would put load on the pump shaft causing amps to increase.
VALID DISTRACTOR: because the applicant may think that the increased casing temperature is due to bearing failure.

CHOICE (B) - NO
WRONG: Runout conditions would be indicated by higher, not lower amps.
VALID DISTRACTOR: because the applicant the abnormal noise is caused by runout conditions.

CHOICE (C) - YES
During SDC warmup, only one LPSI pump should be running to prevent the performance imbalances between the pumps from seating one of the pump's discharge check valves. This causes that pump to run at shutoff head, causing it to overheat.

CHOICE (D) - NO
WRONG: Suction line vortexing is a phenomenon associated with mid-loop operations where air becomes entrained in the SDC suction. Given the RCS is 290 degrees F and 300 psia, conditions do not exist to allow SDC suction line vortexing.
VALID DISTRACTOR: because the applicant may recognize that the low amps are one indication of loss of suction, which can occur when air is entrained in the suction line.

References

1. SDC-00-C, "Shutdown Cooling System", Revision 3 (1/24/03) (Pg 31 of 79)

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.7 **RO** 3.7 **SRO** 4.4 **CFR Link** (CFR: 43.5 / 45.12 / 45.13)

"Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation."

24 ☑ RO ☑ SRO Question ID: 5000016 Origin: Bank ☐ Memory Level

A LOCA Outside of Containment has occurred at the plant. In addition, during post trip EOP actions it was determined that reactor coolant radiation levels are significantly above normal. 15 minutes have elapsed since the reactor trip occurred. A General Emergency Classification has been made by the Shift Manager. The emergency response organization has NOT yet been staffed.

It has been determined that the LOCA can be isolated from the Mechanical Penetration area, however dose rates are very high. Radiation Protection Group surveys indicate that the general area dose rate is 50 REM per hour in the area of the Mechanical Penetration area.

Using Emergency Exposure Limits, what is the MAXIMUM stay time for an operator entering the area to isolate the leak with all dose extensions necessary for this condition granted?

A 6 minutes ☐

B 30 minutes ☑

C 90 minutes ☐

D 120 minutes ☐

Justification

For "protection of large populations" the dose limit utilizing Emergency Exposure limits is 25 Rem. If the Dose Rate is 50 Rem/hr in the vicinity, the stay time would be 30 minutes. A, C, D distractors are equivalent to 5 Rem, 75 Rem, and 100 Rem - All plausible Emer and Non Emer numbers.

CHOICE (A) - NO
WRONG: Dose would be 5 Rem. Limit is 5 times annual non-emergency limit of 5 Rem TEDE, which is equal to 25 Rem.
VALID DISTRACTOR: because 5 Rem, an established dose limit for non-emergency situations, is plausible.

CHOICE (B) - YES
Dose would be 25 Rem, which is the Emergency Exposure limit.

CHOICE (C) - NO
WRONG: Dose would be 75 Rem. Limit is 5 times annual non-emergency limit of 5 Rem TEDE, which is equal to 25 Rem.
VALID DISTRACTOR: because 75 Rem, an established dose limit for life-threatening emergency situations, is plausible.

CHOICE (D) - NO
WRONG: Dose would be 100 Rem. Limit is 5 times annual non-emergency limit of 5 Rem TEDE, which is equal to 25 Rem.
VALID DISTRACTOR: because 100 Rem is plausible.

References

1. 10CFR20 "Standards for Protection Against Radiation", Subpart 20.1206, "Planned Special Exposures"
2. RPM 5.1.5, "Planned Special Exposures"
3. Source: Indian Point 3 NRC Exam, 12/2003

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.4 **RO** 2.5 **SRO** 3.1 **CFR Link** (CFR: 43.4 / 45.10)

"Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized."

25 ☑ RO ☑ SRO Question ID: 5000018 Origin: Bank ☑ Memory Level

A steam generator tube rupture has occurred on SG2. EOP-2534, "Steam Generator Tube Rupture" has been implemented. Which of the following actions is performed in accordance with EOP-2534, "Steam Generator Tube Rupture" to DIRECTLY limit the potential radiation release to the public?

A raise ruptured SG level above 40% using TDAFW pump ☐

B ensuring ruptured SG ADV setpoint at 920 psia and closed ☑

C tripping RCPs if pressurizer press less that 1714 psia and SIAS initiated ☐

D entering EOP-2536 (ESDE) for a SG pressure < 800 psia and subcooling going up ☐

Justification

CHOICE (A) - NO
WRONG: MSI blocked to facilitate controlled cooldown via preferred method (turbine bypass valves to condenser)
VALID DISTRACTOR: because MSI is blocked by procedure.

CHOICE (B) - YES
The ADV is ensured to be in auto and its setpoint is raised to a value below the upper end of the band. It is also ensured to be closed since steam pressure should be below this point. This places the ADV in a condition to open prior to pressure in the isolated steam generator reaching the MSSV lift setpoint, and minimizing the possibility that a MSSV will open and stick in an open position.

CHOICE (C) - NO
WRONG: RCP trip strategy based on worst-case LOCA concerns. Continued operation of pumps is preferable during a SGTR event to allow for a prompt controlled RCS cooldown and depressurization.
VALID DISTRACTOR: because RCP trip is directed by procedure under specified conditions.

CHOICE (D) - NO
WRONG: Functional Recovery Procedure is used to address multiple events from a symptom-based perspective. EOP-2536 should not be entered and implemented from EOP-2534 with multiple events in progress
VALID DISTRACTOR: because EOP-2534 does contain diagnosis confirmation steps and the functional recovery does address excess steam demand events.

References

1. EOP-2534, "Steam Generator Tube Rupture", Revision 22, (3/22/02), (Pg 10, 17 of 64)
2. TG-2534, Steam Generator Tube Rupture, Revision 21 (Pg 14, 22, 30, 37 of 126)
3. Source: Indian Point 3 NRC Exam, 12/2003

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.11 **RO** 2.7 **SRO** 3.2 **CFR Link** (CFR: 45.9 / 45.10)

Ability to control radiation releases.

26

☑ RO ☑ SRO Question ID: 5000019 Origin: Bank ☑ Memory Level

Unit 2 is operating at 100% power. Which of the following Unit 2 activities/events requires direct notification of Unit 3 personnel?

A planned release of Waste Gas Decay Tank T-19A ☐

B entry into 1 hour TS action statement for a RWST boron sample result of 1685 ppm ☐

C manageable steam leak on body of HD-103A, Feedwater Heater 1A Normal Dump Valve ☐

D small oily rag bin fire in turbine building, extinguished within 10 minutes ☑

Justification

CHOICE (A) - NO
WRONG: Notification of Unit 3 not required for planned releases of waste gas decay tanks.
VALID DISTRACTOR: A radioactive discharge from any unit is of general interest to the entire site. The applicant may therefore think that notification of Unit 3 is required.

CHOICE (B) - NO
WRONG: Notification of Unit 3 not required for entry into TS action statements.
VALID DISTRACTOR: Tech Specs require a unit shutdown to COLD SHUTDOWN if RWST boron concentration remains out of spec for greater than 1 hour. A plant shutdown does require a plant announcement per MP-14-OPS-GDL200.

CHOICE (C) - NO
WRONG: Notification of Unit 3 not required for steam leaks on Unit 2. The normal dump valve is isolable.
VALID DISTRACTOR: Steam leak is a concern for personnel safety and continued operation. C-OP-200.4 provides direction for addressing the event but does not require Unit 3 notification. A steam leak requiring a unit shutdown would, however, be announced on plant page.

CHOICE (D) - YES
AOP-2559, "Fire" requires direct notification of Unit 3 for all fires.

References

1. MP-14-OPS-GDL200, "Conduct of Operations", Revision 8 (9/09/04) (Pg 27 of 42)
2. C-OP 200.4, "Response to Significant Plant Leaks", Revision 1 (1/26/96)
3. AOP-2559, "Fire", Revision 7 (3/24/04) (Pg 6 of 34)
4. Heater Drains Print 25203-26004 Sheet 3 of 3 (H-9)
5. TS 3.5.4, Refueling Water Storage Tank
6. Source: Indian Point 3 NRC Exam, 12/2003

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.14 **RO** 2.5 **SRO** 3.3 **CFR Link** (CFR: 43.5 / 45.12)

Knowledge of system status criteria which require the notification of plant personnel.

27 ☑ RO ☑ SRO Question ID: 5000025 Origin: New ☑ Memory Level

The plant is at full power. CEA partial movement testing is in progress per SP-2620A when Reg Group 7 CEA #65 drops to 162 steps. In accordance with Technical Specification 3.1.3.1, reactor power is lowered to less than _______ within 1 hour primarily to reduce the effects on _____________.

A 70%, long term power distributions from xenon redistribution ☑

B 85%, long term power distributions from xenon redistribution ☐

C 70%, available shutdown margin used in accident analyses ☐

D 85%, available shutdown margin used in accident analyses ☐

Justification

CHOICE (A) - YES
Basis for Tech Spec 3.1.3.1 explains that power is lowered to 70% to reduce the xenon redistribution effects on long term power distributions

CHOICE (B) - NO
WRONG: The basis explains that for small misalignments (<20 steps) there is a negligible effect on the available SHUTDOWN MARGIN.
VALID DISTRACTOR: TS 3.10.2, "SPECIAL TEST EXCEPTIONS - GROUP HEIGHT AND INSERTION LIMITS" limits power level to 85%.

CHOICE (C) - NO
WRONG: The basis explains that for small misalignments (<20 steps) there is a negligible effect on the available SHUTDOWN MARGIN.
VALID DISTRACTOR: TS 3.1.3.1 limits power level to 70%. The basis states that the specifications of section 3.1.3.1 ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

CHOICE (D) - NO
WRONG: The basis explains that for small misalignments (<20 steps) there is a negligible effect on the available SHUTDOWN MARGIN.
VALID DISTRACTOR: TS 3.10.2, "SPECIAL TEST EXCEPTIONS - GROUP HEIGHT AND INSERTION LIMITS" limits power level to 85%.

References

1. T.S. 3.1.3.1 "MOVABLE CONTROL ASSEMBLIES - CEA POSITION", Amendment 280 (Pg 3/4 1-20)
2. T.S. 3.1.3.1 Basis (Page B 3/4 1-3a, B 3/4 1-4)

NRC K/A System/E/A

System 005 Inoperable/Stuck Control Rod

Number AK3.05 **RO** 3.4 **SRO** 4.2 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Inoperable / Stuck Control Rod: Power limits on rod misalignment

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

28 ☑ RO ☑ SRO Question ID: 5000026 Origin: New ☐ Memory Level

The plant is in MODE 6 with refueling operations in progress.

One Wide Range Excore Nuclear Instrument has failed; repairs are in progress.

An I&C supervisor calls the control room and reports that, based on an audit of completed surveillances, it has been determined two of the remaining channels were improperly calibrated by an inexperienced technician and should be considered inoperable. The remaining channel was properly calibrated.

What impact does this have on fuel handling activities and why?

A All fuel movement in containment and the spent fuel pool must be suspended due to inadequate remaining instrumentation for monitoring the state of the core. ☐

B All fuel movement in to and out of the reactor core must be suspended due to inadequate remaining instrumentation for monitoring the state of the core. ☑

C Fuel movement may continue since the operability of the remaining channel is adequate for monitoring the state of the core. ☐

D Fuel offload activities may proceed; fuel reload must be suspended due to inadequate remaining instrumentation for monitoring positive reactivity additions. ☐

Justification

CHOICE (A) - NO
WRONG: CORE ALTERATIONS must be suspended without two operable channels, activities in the spent fuel pool are not CORE ALTERATIONS.
VALID DISTRACTOR: Plausible that all fuel handling would be stopped.

CHOICE (B) - YES
CORE ALTERATIONS must be immediately suspended; fuel movement in the core is a subset.

CHOICE (C) - NO
WRONG: Minimum channels operable requirement is TWO source range channels.
VALID DISTRACTOR: Plausible that one channel sufficient for core alterations.

CHOICE (D) - NO
WRONG: All CORE ALTERATIONS must be suspended, not just those involving positive reactivity additions.
VALID DISTRACTOR: Plausible that only concerned about addition of positive from new fuel loading.

Note: This question, on both the RO and SRO exams, samples CFR 55.43(6), "Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity."

References

1. T.S. 3.9.2 /4.9.2, "REFUELING OPERATIONS - INSTRUMENTATION", Amendment 263 (Pg 3/4 9-2, B 3/4 9-1)

NRC K/A System/E/A

System 032 Loss of Source Range Nuclear Instrumentation

Number AK3.02 **RO** 3.7* **SRO** 4.1 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Source Range Nuclear Instrumentation: Guidance contained in EOP for loss of source-range nuclear instrumentation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

29 ☑ RO ☑ SRO Question ID: 0053696 Origin: Bank ☑ Memory Level

Which one of following boration flowpaths would be available after a loss of B-51?

- **A** BAST gravity feed valves and a charging pump. ☐
- **B** BA pump, BA isolation (514) and a charging pump. ☑
- **C** BA pump, BA flow control valve, VCT outlet (501), and a charging pump. ☐
- **D** RWST isolation valve, RWST to chg pp suction (CH-504) and a charging pump. ☐

Justification

CHOICE (A) - NO
WRONG: CH-508 and CH-509 are powered from B-51.
VALID DISTRACTOR: May think that gravity feed flowpath not affected by loss of B-51.

CHOICE (B) - YES
No components in this flowpath are affected by a loss of B-51.

CHOICE (C) - NO
WRONG: CH-501 powered by B-51.
VALID DISTRACTOR: May think that 501 unaffected by loss of B-51.

CHOICE (D) - NO
WRONG: CH-504 powered by B-51.
VALID DISTRACTOR: May think that 504 unaffected by loss of B-51.

References

1. ESA-01-C, "Engineered Safety Features Actuation System" Lesson, Revision 3 (8/6/01)

NRC K/A System/E/A

System 004 Chemical and Volume Control System

Number A4.18 **RO** 4.3 **SRO** 4.1 **CFR Link** (CFR: 41/7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Emergency borate valve

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

30 ☑ RO ☑ SRO Question ID: 5000028 Origin: New ☑ Memory Level

The technical specification allowed outage time for one train of containment spray reflects the dual function of containment spray for _____________.

A heat removal and iodine removal ☑

B heat removal and sump pH control ☐

C hydrogen reduction and iodine removal ☐

D hydrogen reduction and sump pH control ☐

Justification

CHOICE (A) - YES
Per TS Basis: The containment spray is more effective than the containment cooling system in reducing the temperature of superheated steam inside containment following a main steam line break. In addition, the containment spray system provides a mechanism for removing iodine from the containment atmosphere. Therefore, at least one train of containment spray is required to be OPERABLE when pressurizer pressure is >1750 psia, and the allowed outage time for one train of containment spray reflects the dual function of containment spray for heat removal and iodine removal.

CHOICE (B) - NO
WRONG: Sump pH control is provided by trisodium phosphate (TSP) dodecahydrate stored in dissolving baskets located in the containment basement. It functions to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP will get into solution during a LOCA even if containment spray is unavailable. Sump pH control is not a function of containment spray.
VALID DISTRACTOR: Control of pH is provided by TSP.

CHOICE (C) - NO
WRONG: Per lesson material: The introduction of highly acidic borated water in a fine mist to the containment will result in the liberation of hydrogen gas in containment. This is produced as a result of the metal-water reaction with aluminum and zinc components. Corrosion of these components is minimal and therefore the brief exposure to containment spray will result in negligible loss of structural integrity of these components. The generation of hydrogen by this mechanism is minimized by controlling the inventory of susceptible metals and by neutralizing the acidity of the water with Trisodium Phosphate.
VALID DISTRACTOR: Amount of hydrogen generation is minimized, but hydrogen concentration is not reduced, by sump pH control.

CHOICE (D) - NO
WRONG: Per lesson material: The introduction of highly acidic borated water in a fine mist to the containment will result in the liberation of hydrogen gas in containment. This is produced as a result of the metal-water reaction with aluminum and zinc components. Corrosion of these components is minimal and therefore the brief exposure to containment spray will result in negligible loss of structural integrity of these components. The generation of hydrogen by this mechanism is minimized by controlling the inventory of susceptible metals and by neutralizing the acidity of the water with Trisodium Phosphate.
VALID DISTRACTOR: Amount of hydrogen generation is minimized, but hydrogen concentration is not reduced, by sump pH control.

References

1. TS 3/4 6.2.1 Basis, "CONTAINMENT SYSTEMS - DEPRESSURIZATION AND COOLING SYSTEMS - CONTAINMENT SPRAY AND COOLING SYSTEMS", Amendment 236 (Pg B 3/4 6-3)
2. CSS-00-C, "Containment Spray System" Lesson, Revision 4 (1/16/01), Section D.2.a. (Pg 23 of 54)

NRC K/A System/E/A

System 026 Containment Spray System (CSS)

Number K4.06 **RO** 2.8 **SRO** 3.2* **CFR Link** (CFR: 41.7)

Knowledge of CSS design feature(s) and/or interlock(s) which provide for the following: Iodine scavenging via the CSS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

31 ☑ RO ☑ SRO Question ID: 0053994 Origin: Bank ☑ Memory Level

The plant is operating at 100% power with bus 24E aligned to bus 24D. The "B" RBCCW Heat Exchanger is not in service.

The "A" RBCCW Pump breaker trips and the first attempt to remotely reclose the breaker are not successful. A PEO is dispatched to determine why the breaker cannot be closed remotely.

With regard to the RBCCW system, which of the following actions must be performed?

A Align and start the 'B' RBCCW pump to supply Facility 1 RBCCW Header. ☑

B Immediately trip the reactor, then trip the affected RCPs due to the loss of RBCCW. ☐

C Realign Bus 24E to Bus 24C and start the 'B' RBCCW Pump on Facility 1 RBCCW Header. ☐

D Coordinate with PEO for a second attempt to reclose the motor breaker. If RCP seals exceed 250 degrees F, then trip reactor and affected RCPs. ☐

Justification

CHOICE (A) - YES
The 'B' RBCCW pump is available to supply Facility 1 within 5 minutes. The guidance provided in AOP allows utilizing the 'B' pump to supply Facility 1 even though it is electrically aligned to Facility 2.

CHOICE (B) - NO
WRONG: AOP directs compensatory actions. A reactor trip would not be required unless unable to restore flow in a timely fashion.
VALID DISTRACTOR: A sustained loss of one header will require a reactor trip.

CHOICE (C) - NO
WRONG: Insufficient time available to realign the bus power source in accordance with procedure. Realignment of power source is not required by AOP.
VALID DISTRACTOR: When performing routine realignments, Bus 24E would be shifted to Bus 24C

CHOICE (D) - NO
WRONG: AOP specifically allows for only one attempt to restart. Focusing all effort on restart of the affected pump could result in a required reactor trip if unsuccessful. Given that breaker has tripped and cannot be immediately reclosed it is likely there may be a problem with the component.
VALID DISTRACTOR: Applicant may think this is the appropriate action.

References

1. AOP-2564, "Loss of RBCCW", Revision 4 (12/6/02) (Pg 6,10 of 46)

NRC K/A System/E/A

System 008

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.23 **RO** 3.9 **SRO** 4.0 **CFR Link** (CFR: 45.2 / 45.6)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

32 ☑ RO ☑ SRO Question ID: 5000029 Origin: New ☐ Memory Level

The plant is operating at full power with 'B' HPSI Pump OOS for maintenance. A sustained loss of Bus 24D occurs. Coincident with the event, a large break LOCA occurs due to a guillotine shear of #1 hot leg. Assuming Bus 24D CANNOT be reenergized, select the choice which correctly completes the following regarding the impact of the loss of ECCS pumps.

12 hours after the event, a loss of the only available ___________ adversely affect long term core cooling because the remaining ______________.

A HPSI pump would, LPSI pump is NOT preferred for boron precipitation control ☐

B HPSI pump would, LPSI pump could NOT be procedurally realigned for boron precipitation control via hot leg injection ☑

C LPSI pump would NOT, HPSI pump is preferred for boron precipitation control ☐

D LPSI pump would NOT, HPSI pump could be procedurally realigned for boron precipitation control via hot leg injection ☐

Justification

CHOICE (A) - NO
WRONG: LPSI can provide adequate flow.
VALID DISTRACTOR: Plausible that HPSI injection necessary for adequate injection flow.

CHOICE (B) - YES
A single HPSI pump will provide sufficient flow for long term cooling. A LPSI pump could physically be aligned for hot leg injection but the EOPs do not provide procedural guidance for performing this task.

CHOICE (C) - NO
WRONG: Loss of LPSI would have adverse affect because of inability to realign HPSI.
VALID DISTRACTOR: HPSI does provide sufficient core cooling flow.

CHOICE (D) - NO
WRONG: HPSI could not be procedurally realigned for boron precipitation control because it is needed for injection.
VALID DISTRACTOR: HPSI could physically realigned but not iaw procedure.

References

1. ECC-01-C, "Emergency Core Cooling System", Revision 3 (6/28/01) (Pg 11,13 of 25)
2. OP-2541, Appendix 18, "Simultaneous Hot and Cold Leg Injection"

NRC K/A System/E/A

System 006 Emergency Core Cooling System (ECCS)

Number K6.13 **RO** 2.8 **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: Pumps

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

 # 33 ☑ RO ☑ SRO Question ID: 5000030 Origin: Bank ☑ Memory Level

The following Quench Tank parameters are provided:

- Temperature is 150°F
- Level is 52%
- Pressure is 3 psig
- Oxygen concentration is 3.2%

What action is required to restore conditions to normal?

A Lower pressure to less than 1 psig. ☐

B Lower level to less than 45%. ☐

C Lower O2 concentration to less than 2%. ☐

D Lower temperature to less than 120°F. ☑

Justification

CHOICE (A) - NO
WRONG: Pressure is normally maintained between 1 and 5 psig.
VALID DISTRACTOR: Applicant may think pressure is too high.

CHOICE (B) - NO
WRONG: Level is maintained at 50% and must be maintained above 45%.
VALID DISTRACTOR: 45% is the low limit.

CHOICE (C) - NO
WRONG: Concentration is maintained less than 4% oxygen.
VALID DISTRACTOR: Applicant may think concentration must be reduced below 3%.

CHOICE (D) - YES
Temperature is maintained below 120°F.

References

1. OP-2301A, "PDT and Quench Tank Operation", Revision 10 (7/26/04) (Pg 4,6,8,16 of 37)
2. ARP-2590B-207, "QUENCH TANK TEMP HI", Revision 0 (3/4/04)
3. Source: INPO Bank - Q# 19360 - Used at Kewaunee 1, 12/11/2000

NRC K/A System/E/A

System 007 Pressurizer Relief Tank/Quench Tank System (PRTS)

Number A1.03 **RO** 2.6 **SRO** 2.7 **CFR Link** (CFR: 41.5/45.5)

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

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

34 ☑ RO ☑ SRO Question ID: 5000031 Origin: New ☐ Memory Level

Unit 2 was in the process of raising power to 100%. Given the following events and conditions:

Reactor power
- Q Power Channel 'A' = 76.0%
- Q Power Channel 'B' = 73.0%
- Q Power Channel 'C' = 72.0%
- Q Power Channel 'D' = 75.0%

Thermal power = 72.5%
Variable High Power Trip (VHPT) set points were last reset at Q=66.5%

Which one of the following statements correctly describes the effect of these conditions?

A The trips have actuated on channels "A" and "D" and the reactor will trip. ☐

B The pretrip has actuated on only channel "A" and CEA withdrawal motion is inhibited. ☐

C The pretrips have actuated on channels "A" and "D" and CEA withdrawal motion is inhibited. ☑

D The VHPT reset pushbuttons are lit on channel "A" but NO pretrips have actuated in any channel. ☐

Justification

CHOICE (A) - NO
WRONG: AB, AC and AD logic ladders have tripped from the signal in Channel "A" but all other channels remain below the VHP trip setpoint. The reactor will not trip unless another channel exceeds 8.8%.
VALID DISTRACTOR: Applicant may think logic is met for a reactor trip.

CHOICE (B) - NO
WRONG: Channels "A" and "D" have exceeded their pretrip setpoint. 2 of 4 pretrips cuase a CEA withdrawal inhibit and control rods will not move out.
VALID DISTRACTOR: If the applicant thinks that CH "A" is no longer in pretrip because it has already tripped then this could be a plausible answer.

CHOICE (C) - YES
Channels "A" and "D" have exceeded their pretrip setpoint. 2 of 4 pretrips cuase a CEA withdrawal inhibit and control rods will not move out.

CHOICE (D) - NO
WRONG: Channels "A" and "D" have exceeded their pretrip setpoint. 2 of 4 pretrips cuase a CEA withdrawal inhibit and control rods will not move out.
VALID DISTRACTOR: If the applicant does not recall the VHP trip setpoint or thinks that they are continually reset during apower ascension (as they are during a power decrease) and compares channel power to thermal power, this distractor could be selected.

References

1. RPS-01-C, "Reactor Protection System" Lesson, Revision 6 (9/15/00), (Pg 19 of 80 and Figures 7, 20, and 33)

NRC K/A System/E/A

System 012 Reactor Protection System

Number K3.01 **RO** 3.9 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the RPS will have on the following: CRDS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

35 ☑ RO ☑ SRO Question ID: 0053886 Origin: Mod ☐ Memory Level

Given the following conditions:
- 100% reactor power
- Inverter 2 has been isolated in preparation for repairs

The DC input breaker on Inverter 6 is inadvertently opened while hanging the clearance on Inverter 2 .

If a large break LOCA were to occur inside containment with the plant in this configuration which of the following would be an expected condition two minutes after the event? Assume NO operator action.

A 'A' LPSI Pump will NOT be running. ☐

B 'B' LPSI Pump will NOT be running. ☑

C 'C' CAR Cooler Fan will be running in fast speed. ☐

D 'D' CAR Cooler Fan will be running in slow speed. ☐

Justification

CHOICE (A) - NO
WRONG: The LBLOCA will actuate SIAS. Facility 1 ESAS equipment will operate as designed.
VALID DISTRACTOR: 'A' LPSI Pump will be running.

CHOICE (B) - YES
Opening DC input breaker on Inverter 6 with Inverter 2 out will deenergize Vital AC Bus VA20, which will deenergize Facility 2 ESAS Actuation Cabinet. All Facility 2 ESAS associated equipment will be prevented from responding to conditions which would normally result in an actuation. 'B' LPSI will remain stopped until manually started by the operator.

CHOICE (C) - NO
WRONG: The LBLOCA will actuate SIAS. Facility 1 ESAS equipment will operate as designed.
VALID DISTRACTOR: 'A' CAR Fan will shift to slow speed.

CHOICE (D) - NO
WRONG: The LBLOCA will actuate SIAS. Facility 2 ESAS equipment will not receive actuation signals.
VALID DISTRACTOR: 'B' CAR Fan will remain in fast speed.

References

1. LVD-00-C, "125 VDC/120 VAC" Lesson, Revision 5 (Pg 9 of 81)

NRC K/A System/E/A

System 013 Engineered Safety Features Actuation System (ESFAS)

Number A2.04 **RO** 3.6 **SRO** 4.2 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations;
Loss of instrument bus

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

36 ☑ RO ☑ SRO Question ID: 5000033 Origin: New ☐ Memory Level

The plant is operating at 75% power when Main Steam Line Pressure Transmitter PT-4224 for the #2 ADV fails high.

Which of the following describes the response of 'B' SG level to this instrument failure? Assume NO operator action.

A Level will NOT change, the feedwater control system will maintain level constant. ☐

B Level will initially increase, then stabilize to maintain a level equal to the level prior to the failure. ☑

C Level will initially decrease, then stabilize to maintain a level equal to the level prior to the failure. ☐

D Level will initially increase, then stabilize to maintain a level higher than the level prior to the failure. ☐

Justification

CHOICE (A) - NO
WRONG: Level will swell when the ADV on #2 MSL opens. The feedwater control system will restore level to setpoint.
VALID DISTRACTOR: Applicant may not recognize the effect of the instrument failure on the ADV.

CHOICE (B) - YES
Level will swell when the ADV on #2 MSL opens. The feedwater control system will restore level to setpoint.

CHOICE (C) - NO
WRONG: Level will swell when the ADV on #2 MSL opens. The feedwater control system will restore level to setpoint.
VALID DISTRACTOR: Applicant may think that the predominant level effect will be shrink due to additional feedwater when FRV opens in response to steam-feed mismatch.

CHOICE (D) - NO
WRONG: Level will swell when the ADV on #2 MSL opens. The feedwater control system will restore level to setpoint.
VALID DISTRACTOR: Applicant may think the controller will maintain level higher than setpoint.

References

1. MSS-00-C, "Main Steam System" Lesson, Revsion 6 (7/11/01), Page 16, 25 of 69)
2. Millstone Unit 2 FSAR, Revision 21, Section 7.4.7 (Pg 7.4-19)

NRC K/A System/E/A

System 039 Main and Reheat Steam System (MRSS)

Number K1.01 **RO** 3.1 **SRO** 3.2 **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: S/G

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

37 ☑ RO ☑ SRO Question ID: 0070216 Origin: Mod ☐ Memory Level

The Main Steam Isolation Valves will automatically close in response to which one of the following sets of conditions?

A PT-1013A, SG 1 CHANNEL A PRESSURE = 575 psia
PT-1013B, SG 1 CHANNEL B PRESSURE = 570 psia ☐

B PT-1013A, SG 1 CHANNEL A PRESSURE = 564 psia
PT-1023B, SG 2 CHANNEL B PRESSURE = 566 psia ☑

C PT-1023A, SG 2 CHANNEL A PRESSURE = 567 psia
PT-1023B, SG 2 CHANNEL B PRESSURE = 574 psia ☐

D PT-1013B, SG 1 CHANNEL B PRESSURE = 552 psia
PT-1023B, SG 2 CHANNEL B PRESSURE = 555 psia ☐

Justification

CHOICE (A) - NO
WRONG: Only CH B is < 572 psia.
VALID DISTRACTOR: Both channels are on same SG.

CHOICE (B) - YES
Both pressures <572 psia, one is CH A, the other CH B. MSI is generated by 2/4 SG pressure <572 psia on any 2 channels, provided they are not the same letter designation. For example: SG1 CH A and SG2 CH B (one A and one B) is an acceptable combination, whereas SG1 CH A and SG2 CH A (both A's) is not an acceptable combination.

CHOICE (C) - NO
WRONG: Only CH A is < 572 psia.
VALID DISTRACTOR: Different channels on the same SG.

CHOICE (D) - NO
WRONG: Both transmitters have same designation (CH B).
VALID DISTRACTOR: Both pressures are < 572 psia.

References

1. ESA-01-C, "Engineered Safety Features Actuation System" Lesson, Revision 3 (8/6/01), Table H and Figure 5 (Pg 52, 53)

NRC K/A System/E/A

System 039 Main and Reheat Steam System (MRSS)

Number A3.02 **RO** 3.1 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 45.5)

Ability to monitor automatic operation of the MRSS, including: Isolation of the MRSS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

38 ☑ RO ☑ SRO Question ID: 5000034 Origin: Mod ☐ Memory Level

Given the following plant conditions:

- 100% power
- SG levels at setpoint
- Steam flow and feed flow matched
- SG2 Feed Flow Transmitter FT-5269A output fails high

With NO operator actions, which of the following describes the expected plant response?

A SG level lowers to the low level reactor trip. ☑

B SG level lowers, but stabilizes above the low level reactor trip. ☐

C SG level rises to the high level turbine trip. ☐

D SG level rises, but stabilizes below the high level turbine trip. ☐

Justification

CHOICE (A) - YES
Output from feed flow transmitters FT-5269A and FT-5269B on the SG2 feed line are averaged for input to the three-element level control. Failing one transmitter high drives the average high. The control system will respond by closing the FRV. The level signal will slowly act on the steam flow signal to moderate the response. However, the relatively rapid response to the feed flow signal will dominate the level input. Without operator action, level will decrease to the low SG level reactor trip setpoint.

CHOICE (B) - NO
WRONG: Output from feed flow transmitters FT-5269A and FT-5269B on the SG2 feed line are averaged for input to the three-element level control. Failing one transmitter high drives the average high. The control system will respond by closing the FRV. The level signal will slowly act on the steam flow signal to moderate the response. However, the relatively rapid response to the feed flow signal will dominate the level input. Without operator action, level will decrease to the low SG level reactor trip setpoint.
VALID DISTRACTOR: Applicant may think that level signal will prevent level from dropping to the low level trip.

CHOICE (C) - NO
WRONG: SG will lower to the low level trip setpoint.
VALID DISTRACTOR: Applicant may think the higher indicated feed flow will cause SG level to rise to the turbine trip.

CHOICE (D) - NO
WRONG: SG will lower to the low level trip setpoint.
VALID DISTRACTOR: Applicant may think the higher indicated feed flow will cause SG level to rise but stabilize below the high level trip based on input to the control system from the level signal.

References

1. FWC-01-C, "Feedwater Control System", Revision 2 (3/22/04) (Pg 7,8 of 46)
2. Source: INPO Bank - Q# 1942 - Used at Palisades 1, 6/14/1999

NRC K/A System/E/A

System 059 Main Feedwater (MFW) System

Number K4.08 **RO** 2.5 **SRO** 2.7 **CFR Link** (CFR: 41.7)

Knowledge of MFW design feature(s) and/or interlock(s) which provide for the following: Feedwater regulatory valve operation (on basis of steam flow, feed flow mismatch)

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

39 ☑ RO ☑ SRO Question ID: 5000035 Origin: New ☐ Memory Level

The plant is at 100% power. I&C is performing troubleshooting in the Facility 1 Auxiliary Feed Actuation Cabinet. A spurious Facility 1 AFAS is generated.

As a result of automatic actions associated with this event, plant efficiency will ________ and action will be taken to reduce ________.

A decrease, reactor power by inserting CEAs while maintaining turbine load constant ☐

B increase, reactor power by inserting CEAs while maintaining turbine load constant ☐

C decrease, turbine load by adjusting load limit to reduce reactor power ☑

D increase, turbine load by adjusting load limit to reduce reactor power ☐

Justification

CHOICE (A) - NO
WRONG: Inserting CEAs will insert negative reactivity but lowering RCS temperature will counter this effect. Power will remain >100% until turbine load is reduced.
VALID DISTRACTOR: Insertion of CEAs does add negative reactivity and would reduce power in a reactor below the POAH.

CHOICE (B) - NO
WRONG: Additional heat required to raise temperature of AFW entering SGs, resulting in a decrease in plant efficiency.
VALID DISTRACTOR: Main feedwater will be automatically throttled to compensate for the AFW flow. Applicant may think efficiency is improved by the reduction of main feedwater.

CHOICE (C) - YES
Additional heat required to raise temperature of AFW entering SGs, resulting in a decrease in plant efficiency. OP-2204, "Load Changes", requires power to be maintained less than 100%

CHOICE (D) - NO
WRONG: Additional heat required to raise temperature of AFW entering SGs, resulting in a decrease in plant efficiency.
VALID DISTRACTOR: Main feedwater will be automatically throttled to compensate for the AFW flow. Applicant may think efficiency is improved by the reduction of main feedwater.

References

1. OP-2204, "Load Changes", Revision 19 (6/29/04) (Pg 17 of 46)

NRC K/A System/E/A

System 059 Main Feedwater (MFW) System

Number A2.01 **RO** 3.4* **SRO** 3.6* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Feedwater actuation of AFW system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

40 ☑ RO ☑ SRO Question ID: 5000036 Origin: Bank ☑ Memory Level

How is power supplied to 120 VAC Instrument Bus VR21 when the LOAD CONNECTED TO NORMAL (amber) lamp is lit on Transfer Switch RS-2?

A 480 VAC from MCC B41A, rectified, and then inverted to 120 VAC ☐

B 125 VDC from battery, supplied to Bus 201D, then to inverted to 120 VAC ☐

C 480 VAC from MCC B61, then through step-down transformer to 120 VAC ☑

D 480 VAC from MCC B62, inverted to 120 VAC then isolating transformer to 120 VAC ☐

Justification

CHOICE (A) - NO
WRONG: MCC B61 provides normal power to VR21
VALID DISTRACTOR: MCC B41A provides alternate power to VR21

CHOICE (B) - NO
WRONG: MCC B61 provides normal power to VR21
VALID DISTRACTOR: Bus 201D provides power to VA20 through INV 6

CHOICE (C) - YES
MCC B61 provides normal power to VR21

CHOICE (D) - NO
WRONG: MCC B61 provides normal power to VR21
VALID DISTRACTOR: MCC B62 provides emergency power to VR21

References

1. LVD-00-C, "125 VDC/120 VAC", Revision 5 (Pg 10, 33 of 77 and Figure 3)
2. In House Single Line Diagrams 25203-30001 and 25203-30024
3. Source: INPO Bank - Q# 20751 - Used at Braidwood 1, 10/29/2001

NRC K/A System/E/A

System 062 A.C. Electrical Distribution

Number K2.01 **RO** 3.3 **SRO** 3.4 **CFR Link** (CFR: 41.7)

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

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

41 ☑ RO ☑ SRO Question ID: 5000037 Origin: New ☑ Memory Level

Unit 2 tripped from 100% 90 minutes ago. Bus 24C did NOT transfer to the RSST and the "A" Emergency Diesel Generator (EDG) tripped on DIESEL GEN 12U DIFFERENTIAL LOCKOUT. Electrical Maintenance and Generation Test Services found a loose lead in the trip circuitry and believes the trip was caused by the loose lead. The lead has been tightened and all other leads have been verifiied tight. One of the actions required to perform a subsequent manual start of the "A" Diesel Generator is to press the Alarm Reset button on the local Diesel skid panel.

Pressing the Alarm Reset button on the local Diesel skid panel ________________.

A closes a contact in the start circuitry which energizes the DC air start solenoid valves ☐

B resets the Shutdown Relay which allows the governor to admit fuel to the diesel ☑

C energizes the Auto Start Relay which resets the mechanical overspeed trip solenoid ☐

D resets the Differential Lockout Relay which allows the DG output breaker to be closed ☐

Justification

A is incorrect. The air start soleniods are NOT energized when the Alarm Reset button is pressed. If that were the case, the Diesel would start.

B is correct. When the shutdown relay is energized, the Diesel is automatically tripped and athe governor is electronically placed in a zero fuel position. When the Alarm Reset button is pressed, the shutdown relay is reset, which allows the governor to admit fuel to the Diesel on demand.

C is incorrect. The Auto Start Relay does NOT rest the mechanical overspeed trip.

D is incorrect. The Differential Lockout Relay is NOT reset when the Alarm Reset button is pressed.

References

1. EDG-00-C, "Emergency Diesel Generator System" Lesson, Revision 7 (8/27/02) (Pg 75,107,142 of 143)
2. OP-2346A, "Emergency Diesel Generators", Revision 25 (6/15/04) (Pg 17 of 99)

NRC K/A System/E/A

System 064 Emergency Diesel Generators (ED/G)

Number K4.03 **RO** 2.5 **SRO** 3.0 **CFR Link** (CFR: 41.7)

Knowledge of ED/G system design feature(s) and/or inter- lock(s) which provide for the following: Governor valve operation

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

42 ☑ RO ☑ SRO Question ID: 5000032 Origin: Mod ☐ Memory Level

Given the following conditions on Unit 2:

- SBLOCA resulted in a Manual SIAS
- A loss of offsite power occurred coincident with the Manual SIAS
- 4160 Volt Bus 24E de-energized on 86-2 lockout
- 'A' and 'B' EDGs have energized their respective buses

How many CAR fans will be operating?

A 1 ☐

B 2 ☐

C 3 ☐

D 4 ☑

Justification

CHOICE (A) - NO
WRONG: CAR Fans are powered off of 480 Volt Buses B05 and B06. These buses will re-energize from the EDGs.
VALID DISTRACTOR: May assume Bus 24E fault will keep Bus 24C or 24D from re-energizing. May assume only 1 fan will automatically restart on loss of offsite.

CHOICE (B) - NO
WRONG: CAR Fans are powered off of 480 Volt Buses B05 and B06. These buses will re-energize from the EDGs.
VALID DISTRACTOR: May assume Bus 24E fault will keep Bus 24C or 24D from re-energizing. May assume 2 fans will automatically restart on loss of offsite.

CHOICE (C) - NO
WRONG: CAR Fans are powered off of 480 Volt Buses B05 and B06. These buses will re-energize from the EDGs.
VALID DISTRACTOR: May assume Bus 24E fault will keep Bus 24C or 24D from re-energizing. May assume Bus 24E fault will keep Bus 24C or 24D from re-energizing.

CHOICE (D) - YES
Buses B05 and B06 will reenergize from the EDGs. CAR fans will restart in slow speed on sequencer.

References

1. CCS-00-C, "Containment and Containment Systems" Lesson, Revision 8 (11/20/00) (Pg 33 of 83)
2. Source: INPO Bank - Q# 23156 - Used at Salem 1, 11/4/2002

NRC K/A System/E/A

System 022 Containment Cooling System (CCS)

Number K2.01 **RO** 3.0* **SRO** 3.1 **CFR Link** (CFR:41.7)

Knowledge of power supplies to the following: Containment cooling fans

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

43 ☑ RO ☑ SRO Question ID: 0053401 Origin: Bank ☑ Memory Level

The plant is in Mode 4, on the RSST, with bus 24E powered from bus 24D, when the "A" Service Water (SW) Pump breaker shorts internally causing a fault on Bus 24C and tripping the 24C-24G tie breaker.

If the appropriate equipment actuates on the Loss of Normal Power to Bus 24C, which one of the following operator actions is required to prevent further equipment damage?

A Perform a normal shutdown of the "A" EDG. ☐

B Start the "B" SW and RBCCW pumps on the Facility to which they are aligned. ☐

C Align the "A" EDG to the Facility 2 SW header. ☐

D Shutdown the "A" EDG using the Emergency Shutdown push buttons. ☑

Justification

The "A" EDG is running without any cooling water, it should be immmediately tripped to prevent damaging the machine. The fault on Bus 24C will prevent the A EDG breaker from closing as well as no SW pump available to the facility.

CHOICE (A) - NO
WRONG: EOP-2525 requires trip of the EDG.
VALID DISTRACTOR: Procedures and lesson material stress that normal shutdown generally preferable because less stressful to engine.

CHOICE (B) - NO
WRONG: The "B" pumps are aligned to Facility 2.
VALID DISTRACTOR: Start of a standby pump is a logical choice.

CHOICE (C) - NO
WRONG: No procedural guidance provided to allow cross-tie of Facility 2 RBCCW with Facility 1 EDG.
VALID DISTRACTOR: Cross tie physically possible.

CHOICE (D) - YES
EOP directs trip of running EDG on loss of service water.

References

1. EOP-2525, "Standard Post Trip Actions", Revision 20 (2/22/01) (Pg 6 of 26)
2. EOP-2525 Standard Post Trip Actions Technical Guide, Revision 20 (Pg 6 of 38)

NRC K/A System/E/A

System 076 Service Water System (SWS)

Number A2.02 **RO** 2.7 **SRO** 3.1 **CFR Link** (CFR: 41.5 / 43.5 / 45/3 / 45/13)

Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Service water header pressure

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

44 ☑ RO ☑ SRO Question ID: 5000038 Origin: New ☐ Memory Level

The unit is operating at full power when a SGTR occurs. Operators manually trip the plant and initiate SIAS. EOP-2525, "Standard Post-Trip Actions" are performed. While performing Step 2 of EOP-2534, "Steam Generator Tube Rupture", a report is received that a large air leak has been discovered on the Station Air header upstream of Containment Header Isolation 2-SA-42 in the 14 foot Aux Bldg General Area. "C" Instrument Air Compressor F3C has been repaired and is in service.

Under these conditions what design feature will enable the Instrument Air header to remain pressurized?

A Opening of 2-SAS-6, Station Air Cross Tie to Unit 3 ☐

B Closing of 2-IA-642, Instrument Air to Station Air Excess Flow Check ☑

C Auto opening of 2-SA-10.1, Cross Tie from Station Air to Instrument Air ☐

D Auto closing of 2-SA-23.1, Cross Tie from Station Air to Containment Air ☐

Justification

CHOICE (A) - NO
WRONG: Given the location of the line rupture, the Unit 3 cross-tie would supply the leak. The leak cannot be isolated from the Unit 3 air when cross-tied.
VALID DISTRACTOR: Step 32 of EOP-2534 directs alignment of Unit 3 to the Unit 2 Service Air System.

CHOICE (B) - NO
WRONG: Excess flow check valves for the containment service air are are located at service air connections in containment.
VALID DISTRACTOR: Major service and instrument air headers are designed with excess flow check valves which isolate in the event of excessive air flow.

CHOICE (C) - YES
Valve 2-SA-10.1 automatically opens when instrument air pressure drops below 85 psig to supply instrument air from station air. Valve 2-SA-11.1 is interlocked to close when 2-SA-10.1 is open to stop flow from the station air compressor to the station air system. All station air compressor flow is re-directed into the instrument air header. In the given situation the station air system leak will be isolated from the instrument air system.

CHOICE (D) - NO
WRONG: Valve SA-23.1 is located within containment and will not isolate the leak.
VALID DISTRACTOR: Valve SA-23.1 is on the service air to containment line and fails closed.

References

1. ISA-00-C, "Station Air & Instrument Air Systems" Lesson, Revision 6 (Pg 12, 13 of 78)
2. OP-2332A-001, "Station Air System Valve Alignment", Revision 8 (Pg 4 of 7)
3. Piping Diagram 25203-26009, "Instrument and Station Air System", Sheet 8 of 10, Revision 32 (9/27/01)

NRC K/A System/E/A

System 078 Instrument Air System (IAS)

Number K4.02 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.7)

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

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

45 ☑ RO ☑ SRO Question ID: 0073948 Origin: Bank ☑ Memory Level

The 'A' Instrument Air Compressor F3A is properly tagged for electrical troubleshooting. Electrical Maintenance has determined that the MANUAL / OFF / AUTO Control Switch requires replacement. The yellow tag on the control switch must be _____________.

A cleared prior to removal of the switch from the panel ☐

B lifted under a "temporary lift" until the new switch is installed ☐

C removed from the switch and attached beside the switch mounting location ☑

D maintained with the switch that is removed until transferred to the new switch ☐

Justification

CHOICE (A) - NO
WRONG: Yellow tag is not required to be cleared prior to removing switch.
VALID DISTRACTOR: Generally tags must be cleared before manipulating or working on boundary components.

CHOICE (B) - NO
WRONG: Yellow tag is not required to be cleared prior to removing switch.
VALID DISTRACTOR: Plausible that tag would be temporary lifted to allow switch to be replaced since tag is for information purposes only.

CHOICE (C) - YES
The tagging procedure states that "If tagged panel switch must be removed, remove tag from switch and attach near panel hole (yellow only)."

CHOICE (D) - NO
WRONG: Yellow tag is not required to be cleared prior to removing switch.
VALID DISTRACTOR: Plausible that, since tag is for information only, it might be kept with the original switch until new switch installed as a way of maintaining control over the tag.

References

1. WC-2, "Tagging", Revision 6 (5/1/03), Attachment 4, "Tagging Practices" (Pg 55 of 85)

NRC K/A System/E/A

System 078

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.13 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of tagging and clearance procedures.

46 ☑ RO ☑ SRO Question ID: 5000040 Origin: New ☐ Memory Level

Plant is operating in MODE 1 when operators see indications of a rapid rise in containment pressure coincident with lowering SG pressure. The reactor is manually tripped. The crew enters EOP-2525, "Standard Post Trip Actions". While scanning the control boards, the SPO observes the following:

- CIAS ACTUATION SIG CH 1 TRIP alarm actuated
- CIAS ACTUATION SIG CH 2 TRIP alarm actuated
- Containment Sump Pump P-33A stopped
- Containment Sump Pump P-33B running
- 2-SSP-16.1 Containment Drain Sump Isolation Valve open
- 2-SSP-16.2 Containment Drain Sump Isolation Valve closed
- 2-CH-505, RCP Bleedoff Isolation Valve closed
- 2-CH-506, RCP Bleedoff Isolation Valve closed

These conditions indicate ____________________________.

A ESAS Block Relay 24VDC power has failed ☐

B SPO has overriden the ESAS signal to SSP-16.1 ☐

C CTMT PRESS HI coincidence has not been met ☐

D An actuation module for CIAS has failed to actuate ☑

Justification

CHOICE (A) - NO
WRONG: Loss of 24VDC block relay power would affect all Facility 2 CIAS components in same manner. Would not have RCP Bleedoff Isolation 2-CH-506 closed with SSP-16.1 open.
VALID DISTRACTOR: 24VDC block power is associated with the Facility 2 CIAS actuation modules.

CHOICE (B) - NO
WRONG: ESAS signal to SSP-16.1 cannot be overriden.
VALID DISTRACTOR: Many ESAS actuation signals can be overriden from switches on the main control boards.

CHOICE (C) - NO
WRONG: CIAS ACTUATION SIG CH 2 TRIP alarm would not be in if coincidence not made up.
VALID DISTRACTOR: Some of the indications provided are consistent with no CIAS

CHOICE (D) - YES
Abnormal ESF response caused by a failure of ESAS Actuation Module AM-607. The module actuates the following components on a Facility 2 CIAS:
- 2-RC-001, RC Hot Leg Sampling==Close
- 2-LRR-43.1 PDT Pump Discharge Valve==Close
- 2-GR-11.I Waste Gas Surge Tank Inlet Valve==Close
- 2-SSP-16.1 Containment Drain Sump Isolation Valve==Close
- P-33B Containment Drain Sump Pump==Stop
- 2-SI-312 N2 to SI Tanks Shutoff Valve

References

1. ESA-01-C, "Engineered Safety Features Actuation System" Lesson, Revision 3 (8/6/01) (Pg 44 of 73 and Tables 4 and 5)
2, CCS-00-C, "Containment and Containment Systems" Lesson, Revision 8 (11/20/00) (Pg 15 of 83)
3. ARP-2590A-138, "CIAS ACTUATION SIG CH 2 TRIP", Revision 0

NRC K/A System/E/A

System 103 Containment System

Number A4.03 **RO** 2.7* **SRO** 2.7* **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: ESF slave relays

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

47 ☑ RO ☑ SRO Question ID: 0053544 Origin: Bank ☐ Memory Level

An estimated critical position calculation is being performed to startup the reactor 29 hours after a plant trip from 100%. Boron concentration is 955 ppm.

Reference Data:
- Power = 100%
- Xenon = 2.41% delta rho
- Samarium = 0.78% delta rho
- Tavg = 572F
- Burnup = 7,500 MWD/MTU
- Boron = 692 ppm

The moderator temperature coefficient is ___________ and if moderator temperature is maintained during the startup at 2°F below the temperature assumed by the ECP, then the critical rod height will be _______________ than the calculated estimated critical position.

A negative, lower ☑

B positive, lower ☐

C negative, higher ☐

D positive, higher ☐

Justification

CHOICE (A) - YES
MTC is negative at 532°F when boron concentration is below 1400 ppm.

CHOICE (B) - NO
WRONG: MTC is negative at 532°F when boron concentration is below 1400 ppm.
VALID DISTRACTOR: CEA height will be lower.

CHOICE (C) - NO
WRONG: MTC is negative at 532°F when boron concentration is below 1400 ppm.
VALID DISTRACTOR: MTC will be negative.

CHOICE (D) - NO
WRONG: MTC is negative at 532°F when boron concentration is below 1400 ppm.
VALID DISTRACTOR: If MTC was positive, then CEA height would be higher.

References

1. OP-2208-003, "MODERATOR TEMPERATURE COEFFICIENT VERSUS BORON CONCENTRATION MOC CYCLE 16", Revision 044 (4/27/04)

NRC K/A System/E/A

System 001 Control Rod Drive System

Number K5.26 **RO** 3.3 **SRO** 3.6 **CFR Link** (CFR: 41.5/45.7)

Knowledge of the following operational implications as they apply to the CRDS: Definition of moderator temperature coefficient; application to reactor control

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

48 ☑ RO ☑ SRO Question ID: 5000041 Origin: New ☑ Memory Level

The plant is operating at 100% power when a large break LOCA occurs.

If malfunctions prevent the use of either hydrogen recombiner, identify the approximate time that will elapse from the start of the event before hydrogen concentration will reach 3% by volume inside containment.

A 14 days ☑

B 7 days ☐

C 72 hours ☐

D 10 hours ☐

Justification

CHOICE (A) - YES
HCS Lesson material and EOP-2532 Technical Guide both state the time at 12 to 16 days.

CHOICE (B) - NO
WRONG: Hydrogen concentration expected to reach 3% at 12 to 16 days.
VALID DISTRACTOR: Plausible that concentration would be approaching the limit within one week.

CHOICE (C) - NO
WRONG: Hydrogen concentration expected to reach 3% at 12 to 16 days.
VALID DISTRACTOR: Hydrogen monitor tech spec action time limit is 72 hours.

CHOICE (D) - NO
WRONG: Hydrogen concentration expected to reach 3% at 12 to 16 days.
VALID DISTRACTOR: EOP-2532 directs simultaneous hot and cold leg injection at 10 hours.

References

1. EOP-2532 Loss of Coolant Accident Technical Guide, Revision 21 (Pg 5 of 188)
2. HCS-00-C, "Hydrogen Control System" Lesson, Revision 3 (6/29/01) (Pg 44 of 48)

NRC K/A System/E/A

System 028 Hydrogen Recombiner and Purge Control System (HRPS)

Number K3.01 **RO** 3.3 **SRO** 4.0 **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the HRPS will have on the following: Hydrogen concentration in containment

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

49 ☑ RO ☑ SRO Question ID: 5000042 Origin: Mod ☑ Memory Level

The following conditions exist on Unit 2:

- Reactor power is 80%, steady state at EOC
- RCS boron concentration is 135 ppm
- All systems are in automatic control

'A' Main Steam Pressure Instrument PT-4216 output drifts high causing 'A' Steam Dump Valve to Condenser, 2-MS-209, to stroke to approx 30% open.

Assuming NO immediate operator action, what is the expected response of the plant due to the steam dump valve failure AND what action can the operator take from the control room to stop the excess steam flow?

A Turbine load will decrease by approx. 3% AND reactor power will remain constant. The operator can stop dumping excess steam by placing the Bypass to Condenser Controller PIC-4216 to MAN and in the CLOSE position. ☐

B Turbine load will decrease by less than 3% AND reactor power will increase by approx. 3%. The operator can stop dumping excess steam by placing the Bypass to Condenser Controller PIC-4216 to MAN and in the CLOSE position. ☑

C Turbine load will decrease by approx. 3% AND reactor power will remain constant. The operator can stop dumping excess steam by taking the Quick Open Permissive Switch to OFF. ☐

D Turbine load will decrease by less than 3% AND reactor power will increase by approx. 3%. The operator can stop dumping excess steam by taking the Quick Open Permissive Switch to OFF. ☐

Justification

CHOICE (A) - NO
WRONG: Reactor power will increase.
VALID DISTRACTOR: Some steam flow will divert to the condenser. Turbine load will decrease slightly.

CHOICE (B) - YES
Turbine load will decrease slightly due to lowered steam pressure. Reactor power will increase because of greater steam demand. The valve can be closed by taking controller to manual and reducing output.

CHOICE (C) - NO
WRONG: Reactor power will increase.
VALID DISTRACTOR: Turbine load will decrease slightly.

CHOICE (D) - NO
WRONG: Quick Open Permissive Switch will not close the valve.
VALID DISTRACTOR: Turbine load will decrease slightly and reactor power will increase. The quick open permissive blocks quick open to the atmospherics.

References

1. MSS-00-C, "Main Steam System" Lesson, Revision 6 (7/11/01) (Pg 34, 35 of 74)
2. RRS-01-C, "Reactor Regulating System" Lesson, Revision 3 (7/2/01)
3. ARP-2590D-024, "CONDENSER BYPASS VALVE NOT CLOSED", Revision 0 (2/12/04)
4. Piping Diagram 25203-26002, "Main Steam Turbine", Sheet 4 of 5, Revision 20 (9/17/01) (J-2)
5. Source: INPO Bank - Q# 21444 - Used at Braidwood 1, 7/17/2002

NRC K/A System/E/A

System 045 Main Turbine Generator (MT/G) System

Number A2.08 **RO** 2.8 **SRO** 3.1* **CFR Link** (CFR: 41.5/43.5/45.3/45.5)

Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control,or mitigate the consequences of those malfunctions or operations:
Steam dumps are not cycling properly at low load, or stick open at higher load (isolate and use atmospheric reliefs when necessary)

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

50 ☑ RO ☑ SRO Question ID: 5000039 Origin: New ☐ Memory Level

The plant is at 85% power. Heater Drain Pump 'A' has been removed from service for maintenance on the pump. Given the following tagout boundaries, identify the correct component operation sequence to prevent overpressurization of piping.

1. CLOSE 'A' Heater Drains Pump Suction Valve 2-HD-7A
2. CLOSE 'A' Heater Drains Pump Minimum Flow Recirc Isolation 2-HD-45A
3. CLOSE 'A' Heater Drains Pump Discharge 2-HD-9A

A 1-3-2 ☐

B 3-1-2 ☑

C 1-2-3 ☐

D 2-1-3 ☐

Justification

CHOICE (A) - NO
WRONG: Suction valve is closed before discharge valve.
VALID DISTRACTOR: Procedure directs discharge before recirc.

CHOICE (B) - YES
Discharge valve must always be closed before suction valve to prevent overpressurization of suction piping

CHOICE (C) - NO
WRONG: Suction valve is closed before discharge valve.
VALID DISTRACTOR: Procedure directs drain after isolations.

CHOICE (D) - NO
WRONG: Suction valve is closed before discharge valve.
VALID DISTRACTOR: This sequence would isolate and depressurize piping.

References

1. OP-2320, "Feedwater Heater Drains and Vents", Revision 16 (12/23/03), Section 4.5 (Pg 21 of 46)

NRC K/A System/E/A

System Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.13 **RO** 3.6 **SRO** 3.8 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of tagging and clearance procedures.

51 ☑ RO ☑ SRO Question ID: 0054226 Origin: Mod ☐ Memory Level

The plant was operating at 100% when a reactor trip occurred. Given the following conditions and events:

- 2 charging pumps are operating.
- 3 CEAs failed to insert.
- 2-CH-514 will NOT open.
- 2-CH-508 and 509 will NOT open.

Which one of the following statements correctly describes the procedure and required actions to be taken?

A Continue EOP-2525, "Standard Post Trip Actions" to determine if any other problems exist. Maintain Tavg at or above 500°F ☐

B Refer to AOP-2558, "Emergency Boration" and emergency borate for at least 2 hours by opening 2CH-195 and 2CH-210 to bypass around 2CH-514. ☐

C Continue EOP-2541 "Appendix 3 - Emergency Boration" and emergency borate from the RWST. ☑

D Refer to EOP-2540A, "Functional Recovery of Reactivity Control" and emergency borate using success path RC-3 (Boration using SI). ☐

Justification

EOP 2525 contingency for Reactivity Control

CHOICE (A) - NO. If the candidate thinks that stuck 3 CEAs will not pose a problem as long as Tavg remains above 500 F, this answer is plausible. However, EOP 2525 directs the operator to emergency borate using EOP 2541 appendix 3.

CHOICE (B) - NO. If the candidate thinks that refering to the AOP is permissible under these cricumstances, the flow path will provide boric acid flow to the RCS. The thumbrule requirement in the AOP is to borate 1.5 hours for each additional CEA stuck beyond 1. In this case, if the AOP was used, the requirement to borate would be 3 hours not 2 hours. The applicant would select 2 hours if they used the thumbrule for 3 CEAs stuck out and neglected to recall that one of the CEAs is already considered stuck out in the saefty analysis. Note that an entry condition for AOP 2558 is

CHOICE [C] - YES. Correct answer. EOP 2525 directs the applicant to emergency borate using EOP 2541 appendix 3. There is no time limit provided under emergency conditions. This boration path uses the charging pumps for injection.

CHOICE [D] - NO. If the candidate thinks that a failure of the CVCS primary emergency boration flow path is justification for refering to EOP 2540A, they could possibly select this answer. This is the correct success path if RC-1 (CEA Insertion) and RC-2 (Boration using CVCS) are not available. This success path uses the SI pumps not the charging pumps. Note that RC-2 directs the operator to borate using EOP 2541 Appendix 3.

References

1. AOP-2558 pages 3-9
2. EOP-2525 page 3
3. EOP-2541 Appendix 3 pages 1-5
4. EOP-2540A pages 8-12

NRC K/A System/E/A

System 029

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.23 **RO** 3.9 **SRO** 4.0 **CFR Link** (CFR: 45.2 / 45.6)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

52 ☑ RO ☑ SRO Question ID: 5000056 Origin: New ☐ Memory Level

Unit 2 was operating at 100% power when a spurious reactor trip occurred. Given the following events and conditions:

- Pressurizer level rapidly dropped to 20%
- The operators reach step 3b of EOP-2525 (Standard Post Trip Actions) "Determine Status of RCS Inventory Control"

What is the reason for verifying that RCS subcooling is greater than or equal to +30°F in step 3b?

A To ensure that pressurizer level indication accurately represents total RCS inventory ☑

B To ensure that RCS pressure is under automatic control and returning to the normal band ☐

C To determine if manual control of the pressurizer level control system is required ☐

D To determine if a LOCA is in progress and the contingency action is required ☐

Justification

CHOICE [A] - YES
This is the correct reason for validating that subcooling is > 30°F in this step.

CHOICE [B] - NO
The PLCS does not control pressurizer pressure. RCS pressure control is verified in step 4 of EOP 2525

CHOICE[C] - NO
This distracter is the correct answer for step 3a not step 3b.

CHOICE [D]
This is a correct reason for monitoring RCS subcooling during other steps in the EOPs. There is no contigency action for step 3b.

References

1. EOP-2525 page 8
2. EOP Tech Guide 2525 pages 10-14

NRC K/A System/E/A

System 028 Pressurizer (PZR) Level Control Malfunction

Number AK3.05 **RO** 3.7 **SRO** 4.1 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions:
Actions contained in EOP for PZR level malfunction

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

53 ☑ RO ☑ SRO Question ID: 5000057 Origin: New ☐ Memory Level

Unit 2 was operating at 100% power when a large break LOCA (design bases) occurred. Given the following events and conditions:

- The SDC HX "A" RBCCW outlet valve (2-RB-13.1A) jammed shut and will not open on SRAS
- The Unit Substation Transformer 24D7-1X secondary windings open due to a fault and bus 22E is deenergized

Which one of the following statements correctly decribes the effect of this failure on the containment spray system if repairs CANNOT be made?

A The initial containment pressure spike will exceed design pressure and long term heat removal will NOT be adequate ☐

B The initial containment pressure spike will exceed design pressure but long term heat removal will be adequate ☐

C The initial containment pressure spike will not exceed design pressure but long term heat removal will NOT be adequate ☐

D The initial containment pressure spike will not exceed design pressure and long term heat removal will be adequate ☑

Justification

The loss of SDC to the A train of containment spray will prevent that train from removing heat during long term sump recirc operations. The initial pressure spike will be mitigated because both trains of CSS will inject from the RWST during the initial pressurization of containment. Long term cooling requires operation of one complete train of ESF equipment which includes 2 CAR fans plus one CSS train. Loss of the 22E emergency bus will cause CAR fans F14A and F14C to lose power. CAR fans F14B and F14D will run from bus 22F.

CHOICE [A] - NO
Both trains of containment spray will inject and maintain the initial pressure spike below the design threshold for containment. Long term heat removal requires only 1 train of containment spray and 2 CAR fans to remove sufficient heat.

CHOICE [B] - NO
Both trains of containment spray will inject and maintain the initial pressure spike below the design threshold for containment. Long term heat removal will be adequate with 2 CAR fans and train B containment spray.

CHOICE [C] - NO
Long term heat removal from containment will be adequate with 2 CAR fans and Train B of CSS.

CHOICE [D] - YES

References

1. ECC-01-C rev 3 page 19
2. ECC01 figure 6 (M2105-11-98)
3. CSS-00-C rev 4 chg 1 pages 6-8, 37
4. FSAR Figure 25203-3001 Main Single Line Diagram 1-10-92
5. RBC-00-C rev 5 pages 24-25

NRC K/A System/E/A

System 005 Residual Heat Removal System (RHRS)

Number K3.06 **RO** 3.1* **SRO** 3.2* **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunction of the RHRS will have on the following: CSS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

54 ☑ RO ☑ SRO Question ID: 3 Origin: Bank ☐ Memory Level

The plant is at 15% power with a startup and power ascension in progress when the in-service pressurizer pressure controller is inadvertently shifted to manual. All other pressurizer control components are operating normally for 15% power.

Which one of the following describes the expected effect on the Pressurizer Control System as a result of the pressure controller now being in manual? (Assume NO operator action.)

A Spray valves will begin to open if RCS pressure should begin to rise. ☐

B Backup heaters can ONLY be energized or deenergized manually. ☐

C Low Pressurizer Pressure alarm will NOT actuate regardless of how low pressure drops. ☑

D Proportional heater output will rise ONLY if pressurizer level rises to ~4% above program level ☐

Justification

A - wrong; With the controller in manual, spray valves will NOT operate.

B - wrong; BU htrs are controlled by bistables not the controller, therefore will operate at the bistable setpoints regardless of the controller condition.

C - wrong; alarm is controlled by bistables, not controller.

D - Correct; Prop htr controlled by Controller only, except on level insurge >/= 3.6% above setpoint.

References

1. PLC-01-C.R3 pages 9-12, 21-24, 27-28, 30

NRC K/A System/E/A

System 010 zer Pressure Control System (PZR PCS)

Number A3.02 **RO** 3.6 **SRO** .35 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the PZR PCS, including: PZR pressure

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

55 ☑ RO ☑ SRO Question ID: 5000059 Origin: New ☐ Memory Level

Unit 2 was operating at 100% power when a reactor trip occurred due to a loss of main feedwater. Given the following events and conditions:

- A and B S/G levels dropped to 15%
- The Turbine driven AFW pump was started
- Vital DC bus DV10 was deenergized due to a fault

Which one of the following statement correctly describes the status of:
1. the turbine driven AFW pump, and
2. the AFW FRVs?

A 1. The turbine driven AFW pump will trip and CANNOT be restarted from the control room ☐
2. FW-43A fails open, FW-43B is unaffected

B 1. The turbine driven AFW pump CANNOT be controlled from the control room but can be controlled locally ☐
2. FW-43A and FW-43B fail open

C 1. The turbine driven AFW pump can be controlled by selecting DV20 using the key lock isolation switches on C-05 ☐
2. FW-43A and FW-43B fail open

D 1. The turbine driven AFW pump can be controlled by selecting DV20 using the key lock isolation switches on C-05 ☑
2. FW-43A fails open, FW-43B is unaffected

Justification

The TD AFW pump can be started if vital DC power from DV20 is selected using the key lock switches on C-5. FW-43A fails open on a loss of DC power from DV10. FW-43B will not lose power and will operate normally.

CHOICE [A] - NO
WRONG This answer is partially correct. The TD AFW pump will not trip. This choice is provided if the candidate does not recognize that the TD AFW pump has an alternate vital DC supply that is slected on C-5. Part 2 of this answer is correct.

CHOICE [B] - NO
WRONG The TD AFW pump can be started by realigning DC power from DV20 on C-5. FW-43B Is not effected by the loss of DV10.

CHOICE [C] - NO
WRONG This answer is also partially correct. Control power to the TD AFW can be swapped to DV20 from C-5. FW-43B is powered from DV20.

CHOICE [D] - YES
CORRECT This is the correct answer. FW-43B is not powered from DV-10.

References

1. AFW-00-C rev 5 chg 3 pages 14 and 19
2. EOP-2525 rev 20 page 16
3. AOP-2506A rev 3 page 23

NRC K/A System/E/A

System 061 Auxiliary / Emergency Feedwater (AFW) System

Number K6.01 **RO** 2.5 **SRO** 2.8* **CFR Link** (CFR: 41.7 / 45.7)

Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

56 ☑ RO ☑ SRO Question ID: 0065167 Origin: Mod ☐ Memory Level

Unit 2 was operating at 100% power when an electrical transient occurred. Given the following conditions and events in sequence:

- VA-20 was deenergized
- The plant tripped
- MSI actuated
- A SGTR occurred on the #1 Steam Generator (SG)
- Upon reaching step 6 of EOP 2525 (SPTA) the SPO was directed to feed the #2 SG using Aux Feed Water (AFW)

Which one of the following statements correctly describes:
1. the required actions, and
2. the correct procedure to be used.

A 1. Actions: Place both Facility hand switches in "Pull-To-Lock" - Close 2-FW-44 - Feed #2 SG with the turbine driven AFW pump only.
2. Implement EOP Appendix 6 (TDAFW Pump Normal Startup) ☐

B 1. Actions: Manually initiate facility 2 AFW components - Close 2-FW-44 - Feed #2 SG with the turbine driven AFW pump only.
2. Implement EOP Appendix 7 (TDAFW Pump Abnormal Startup) ☐

C 1. Actions: Place Facility Two hand switch in "Pull-To-Lock" - Ensure #1 AFW Reg. Valve is closed - Locally control #2 Aux FRV.
2. Implement EOP Appendix 6 (TDAFW Pump Normal Startup) ☐

D 1. Actions: Manually start both MDAFW pumps - Ensure #1 AFW Reg. Valve is closed - Locally control #2 Aux FRV.
2. Complete EOP 2525 step 6 without starting the TDAFW pump ☑

Justification

VA-20 powers the actuation logic for facility 2 AFAS and the actuation relays are energize-to-actuate. Loss of VA-20 means that facility 2 AFW components will have to be manually operated. The turbine driven AFW pump should not be used if a SGTR is in progress to prevent radiological contamination. The correct answer is to NOT start the TD AFW pump and close 2-FW-43A (AFW FRV to the #1 S/G) to prevent feeding the ruptured S/G. #2 S/G should be fed using both electric AFW pumps only.

Bank question 0065167 asked the applicants what the correct sequence would be if VA-10 was lost. This question was modified from losing VA-10 to losing VA-20. In addition, the previous question appeared to assume that a loss of VA-10 would fail open the FRV to the #1 S/G. This is not correct - loss of DV-10 causes 2-FW-43A to fail open. This modified question uses the previous bank question but corrects the earlier problems with that revision. Variations of the original distracters are used in the event that appllicants memorized the answer to the bank question.

CHOICE [A] - NO
WRONG This was the previously correct answer to question 0065167 in the MP-2 bank - which was written as a loss of VA-10 instead of VA-20. It is not clear if this answer was ever truly correct. However, this answer is provided as a valid distracter for applicants who may have memorized the bank question. Using the turbine driven AFW pump to feed the #2 S/G when a SGTR is occuring is not recommended when both electric driven AFW pumps are fully functional. Selection of appendix 6 would be appropriate for starting the TDAFW pump and is consistent with the first part of the answer. .

CHOICE [B] - NO
WRONG Although this would result in feeding the #2 S/G, there would be no reason to manually initiate facility 2 AFW components if 2-FW-44 (AFW header cross-connect) was closed. In addition, using the TD AFW pump during a SGTR is not recommended. If the applicant thought that the loss of VA-20 would prevent a normal start of the TDAFW pump, then use of appendix 7 would be correct.

CHOICE [C] - NO
WRONG This distracter is incorrect because there is no reason to place the facility 2 hand switch in pull to lock and feeding the #2 S/G with the TDAFW pump would cause radiological problems - i.e. a release to the environment. Part 1 was an original distracter from the rev 1 version of this question. Use of appendix 6 would be appropriate if the TDAFW did not lose control power - which it does not with a loss of VA-20.

CHOICE [D] - YES
CORRECT The #1 AFW Reg valve (2-FW-43A) remains fully functional despite a loss of VA-20. This valve would fail open if DV10 was lost - which appears to be the previous correct answer to the bank question. Facility 2 AFW components would have to be manually operated because their actuation relay was deenergized when VA-20 lost power.

References

1. AFW-00-C rev 5 chg 3 pages 33 and 35
2. EOP 2525 rev 20 page 16
3. AFW-00-C Figures 1 and 2

NRC K/A System/E/A

System 061 Auxiliary / Emergency Feedwater (AFW) System

Number A2.05 **RO** 3.1* **SRO** 3.4* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Automatic control malfunction

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

57 ☑ RO ☑ SRO Question ID: 5000060 Origin: New ☑ Memory Level

Which one of the following conditions will automatically terminate a liquid release from the AWMT (does NOT require manual operator action to stop the discharge)?

A Reduction in CW flow below authorized dilution flow limit in discharge permit ☐

B Increase in AWMT pump discharge pressure above PIOPs setpoint limit ☐

C Reduction in AWMT pump discharge flow below PIOPs setpoint limit ☑

D Loss of power to the discharge flow recorder ☐

Justification

PIOPs monitors the discharge flow rate and will automatically terminate the discharge if the flow rate exceeds a high or low setpoint.

WRONG (A) - NO
CW flow provides the dilution flow for the discharge. The required amout of dilution flow is stated on the discharge permit. This flow is not monitored by PIOPs and requires operator action to terminate the discharge if it is exceeded.

WRONG [B] - NO
Disharge pressure is monitored in PIOPs but only a low discharge pressure will cause an automatic isolation of the discharge path. A high pressure does not automatically stop the discharge.

CORRECT (C) - YES
Discharge flow rate is monitored by PIOPs and low flow will automatically terminate the discharge.

WRONG [D] - NO
A loss of power to the flow recorder requires operator action to terminate the discharge. There is no automatic action associated with a loss of power to the flow recorder.

References

1. ALR-04-C rev 3 page 10
2. RLD-04-C rev 1 pages 6, 18

NRC K/A System/E/A

System 059 Accidental Liquid Radwaste Release

Number AA2.06 **RO** 3.5* **SRO** 3.8 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: That the flow rate of the liquid being released is less than or equal to that specified on the release permit

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

58 ☑ RO ☑ SRO Question ID: 0054038 Origin: Bank ☐ Memory Level

Unit 2 was operating at 100% power. Given the following events and conditions:

- SP-2601D (Power Range Safety Channel and Delta-T Power Channel Calibration Surveillance Procedure) is in progress
- Reactor Protection System (RPS) Channel 'A' High Power, TM/LP and LPD trip units are bypassed
- RPS Channel 'C' linear power range drawer high voltage power supply fails

Which one of the following statements correctly describes the expected plant response based on the stated conditions with NO further operator actions?

A The plant will trip if EITHER CH 'B' or 'D' processes a trip that receives an input from the linear power range NIs. ☑

B The plant will trip immediately because the 2 out of 4 coincidence logic has been satisfied. ☐

C The plant will trip ONLY if BOTH CH 'B' and 'D' process trips that receive an input from the linear power range NIs. ☐

D A partial trip will be processed because of the opening of ONLY two reactor trip circuit breakers. ☐

Justification

CHOICE [A] - YES
CORRECT Module RPS-01-C, section II.D.7.c states that channel bypass keys operate relays that bypass the trip unit relay contacts. Thus, Channel "A" in the above question does not contribute to 2/4 coincidence logic. OP 2380, Rev. 8, Section 4.4, Part B.3.d.2.c states that a high voltage power failure will cause a single channel trip (Channel "C"). Thus, only one more channel trip ("B" or "D") is necessary to meet 2/4 coincidence logic.

CHOICE [B] - NO
WRONG With channel A bypassed, the logic ladder is in 2/3 channels. A trip in channel C provides only 1 of 2 process inputs required.

CHOICE [C] - NO
WRONG Channel A has input 1 trip signal. All that is required is one more trip signal, not two more trip signals.

CHOICE [D] - NO
WRONG A partial trip will not occur with channel A bypassed.

References

1. RPS-01-CR6 page 14 - 16

NRC K/A System/E/A

System 012 Reactor Protection System

Number A3.02 **RO** 3.6 **SRO** 3.6 **CFR Link** (CFR: 41.7 / 45.5)

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

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

59 ☑ RO ☑ SRO Question ID: 5000061 Origin: New ☑ Memory Level

Unit 2 is shutdown in a refueling outage with maintenance being conducted on DC Bus 201A. Given the following events and conditions:

- "125 VDC LOAD CENTER 201A TROUBLE" annunciator lit (C-08, A-21)
- Battery DB1-201A voltage = 125 volts
- DC Bus 201A voltage = 133 volts
- Charger 201A/DC1 output voltage = 134 volts

Which one of the following statements correctly describes the cause of this problem?

A The PEO has taken the DS-1 open pole detection circuit to the FU-A position ☐

B Battery charger 201A/DC1 output has failed ☐

C A ground has developed on 125 VDC bus 201A (D01) ☐

D Battery disconnect DS-1 has been opened ☑

Justification

CHOICE [A] - WRONG
The DS-1 open pole detection circuit test swtich will not cause this annunciator to actuate. This test swtich is operated routinely by PEOs to verify battery voltage is equal to DC bus voltage. This switch would be used to verify the voltage difference between the battery and the battery bus.

CHOICE [B] - WRONG
Although annuciator C-08, A-21 would actuate, the DC voltage output from the battery charger would not equal 134 VDC if the charger had failed. This could be selected if the applicant though that the charger output would remain at 134 VDC if the charger failed sure to location of the tap for the voltage output.

CHOICE [C] - WRONG
A ground on the DC bus would not actuate C-08, A-21. It would cause a different alarm to actuate. This could be selected if the applicant did not understand the grtound detection circuit.

CHOICE [D] - CORRECT
Opening battery disconnect DS-1 would cause this alarm if battery voltage was greater than 6 volts different between the battery and bus 201A. Battery voltage is 8 volts lower than bus voltage.

Objective: LVD-00-C.R5 PEO Obj 9:
State the purpose and describe the operating characteristics including automatic functions and interlocks, of the following major 125 VDC Electrical Distribution System components as given in LVD-00-C: (MB-00039)
A) Battery Charger
B) Battery
C) Battery Disconnect
D) Battery Fuses
E) Ground Detector
F) 125 VDC Breakers
G) Open Pole Detector
H) Kirk Key Interlocks

References

1. LVD-00-C.R5 CH1 pages 13-14
2. M2104-09-98 87000443 Rev 1 (Figure 2)
3. ARP-2590F Rev 7-04 annuciator C-08, A-21

NRC K/A System/E/A

System 063 DC Electrical Distribution System

Number A3.01 **RO** 2.7 **SRO** 3.1 **CFR Link** (CFR: 41.7 / 45.5)

Ability to monitor automatic operation of the DC electrical system, including: Meters, annunciators, dials, recorders, and indicating lights

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

 # 60 ☑ RO ☑ SRO Question ID: 0054057 Origin: Mod ☐ Memory Level

Unit 2 is operating at 100% power when a plant trip occurs concurrent with a loss of the RSST.

Given the following events and conditions:
- Instrument Air pressure drops slowly to 89 psig immediately following the trip and does not recover

Which one of the following statements correctly describes:
1. The preferred method for restoring Unit 2 Instrument Air, and
2. The reason for selecting this method?

A 1. Cross-tie Unit 2 instrument air with Unit 3 instrument air ☐
2. To maintain air purity in the instrument air system and prevent introduction of contaminants

B 1. Cross-tie Unit 2 station air with Unit 3 service air, then cross-tie Unit 2 instrument air with station air ☑
2. This provides the simplest and fastest way to repressurize instrument air

C 1. Cross-tie Unit 2 instrument air with Unit 2 station air ☐
2. This provides the simplest and fastest way to repressurize instrument air

D 1. Cross-tie Unit 2 station air with Unit 3 service air, then cross-tie Unit 2 instrument air with station air ☐
2. To maintain air purity in the instrument air system and prevent introduction of contaminants

---

Justification

CHOICE [A] WRONG
There is no direct physical connection between the Unit 2 and Unit 3 instrument air systems. In order to cross-tie with Unit 3, the operators must cross-tie Unit 2 station air with Unit 3 service air - then corss-tie Unit 2 station air with Unit 2 instrument air. However, if there was a direct connection, the reason would be valid.

CHOICE [B] - CORRECT
EOP 2525 step 19 requires cross tying Unit 2 to Unit 3 instrument air systems and cross tying unit 2 station air to instrument air. This is the simplest and fastest method of repressurizing instrument air - but it introduces contaminants into the system because service air and station air do not have the same air quality requirement as instrument air.

CHOICE [C] - WRONG
Unit 2 station and instrument air systems are automatically cross-tied whenever instrument air pressure drops below 85 psig and can be manually cross-tied if needed but the Unit 2 station air compressor will not have power if the RSST fails - powered from bus 22C. The reason is valid IF the station air compressor had power.

CHOICE [D] - WRONG
Partially correct - cross tying with unit 3 is the correct action to take, but the station air system does not have the same level of air purity associated with instrument air. The reason - "to maintain air purity..." is incorrect. ISA-00-C lesson material contains a discussion of an occurrence under CR M2-97-2526, where operations cross-tied for approx 6 hours. Based on this occurrence, the operating procedure (OP-2332A) was modified to include a caution regarding minimizing cross-tie duration to prevent moisture buildup in the containment portion of the instrument air system.

References

1. EOP 2525 rev 20 step 19, Contingency Actions for loss of IA.
2. EOP 2528, Step 9 Contingency Actions.
3. ISA-00-C Rev 6 pages 11-16, 51
4. FPS-04-C rev 3 chg 1 page 16
5. Instrument / Station Air Figure 1 M2 10/07/02 87001262
6. EOP 2525 Technical Guide page 35 step 19

NRC K/A System/E/A

System 065 Loss of Instrument Air

Number AK3.04 **RO** 3.0 **SRO** 3.2 **CFR Link** (CFR 41.5,41.10 / 45.6 / 45.13)

Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Cross-over to backup air supplies

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

61 ☑ RO ☑ SRO Question ID: 0054834 Origin: Mod ☑ Memory Level

Unit 2 is shutdown in MODE 6 moving spent fuel from the containment to the spent fuel pool.

Given the following events and conditions:
- Containment purge is in operation
- Spent Fuel Pool and Radwaste Ventilation systems are aligned to exhaust to the Unit 2 stack.
- RM-8123B (Containment Gaseous Activity) alarms due to high activity

Which one of the following changes will occur automatically within the Containment Purge and/or the Main Exhaust systems in response to this condition?

A Supply fan 20 trips and Fuel Handing Area Isolation Valves 2-HV-165, 2-HV-170 and 2-HV-171 close ☐

B Purge supply fan (F-23) trips ☐

C Containment Purge Isolation Valves 2-AC-4, 2-AC-5, 2-AC-6 and 2-AC-7 close ☑

D Main exhaust system makeup air damper (2-AC-59) closes ☐

Justification

CHOICE [A] - WRONG
Main exhaust fans do not trip if RM-8123B alarms. This occurs if an AEAS actuates in the spent fuel pool area. .

CHOICE [B] - WRONG
Purge supply fan F-23 does not trip. If it did trip, it would stop or substantially reduce the flow of containment air through the purge exhaust system.

CHOICE [C] - CORRECT
These purge exhaust valves will close. This correct answer was modified from the original bank question.

CHOICE [D] - WRONG
The bank question answer was that 2-AC-59 OPENED - which is correct. The old right answer was changed to 2-AC-59 CLOSES. It is plausible to expect supply valves to close when a purge must be stopped - but this valve is in the main exhaust system not the purge system.

References

1. RMS-00-C Rev 6 pages 30-33
2. RWV-00-C Rev 4 page 8, 20-22

NRC K/A System/E/A

System 073 Process Radiation Monitoring

Number K4.01 **RO** 4.0 **SRO** 4.3 **CFR Link** (CFR: 41.7)

Knowledge of design feature(s) and/or interlocks which provide for the following: Release termination when radiation exceeds setpoint

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

62 ☑ RO ☑ SRO Question ID: 0053371 Origin: Mod ☑ Memory Level

The plant is operating at 100% power when you receive a high radiation alarm on RM-8132A (Unit 2 Stack Particlulate).

Given the following events and conditions:
- When checked on the PPC and RC-14, RM-8132A is found to be reading 7.5E04 cpm and stable
- An air sample by Health Physics confirms that this is a valid alarm
- HP recommends revising the RM-8132A module setpoint(s) on RC-14

Which one of the following describes what should be done with the RM-8132A module setpoints?

A Raise the alarm setpoint
Do NOT change the alert or fail setpoints ☐

B Raise the alarm setpoint and the alert setpoint
Do NOT change the fail setpoint ☐

C Raise the alarm, alert, and fail setpoints ☑

D Do NOT change the alarm setpoints ☐

Justification

RM-8132A is a process monitor listed in Att 2 of OP2383C Rev 12-02. Raising the setpoints is allowed for all monitors in Att 2.

CHOICE [A] - WRONG
The alarm setpoint must be raised by 2x not 1.5x. The alert and fail setpoints must also be raised.

CHOICE [B] - WRONG
The fail setpoint must also be raised to 1/5 of the new average reading

CHOICE [C] - CORRECT
RM-8132A is a process monitor listed in Att 2 of OP2383C Rev 12-02. Raising the setpoints is allowed for all monitors in Att 2. The fail setpoint is raised to 1/5 of 7.75E04. The alert and alarm setpoints are raised to 1.5 and 2.0 x the new average value

CHOICE [D] - WRONG
If RM-8132A was listed under Att 1 of OP2383C, this would be the correct answer. This was previously the correct answer to the bank question which tested RM-8123B under similar circumstances. This would also be a correct answer if the plant was undergoing a transient.

References

1. OP2383C, "RADIATION MONITOR ALARM SETPOINT CONTROL", Revision 12-02 (08/26/04) (Pg 3, 17-19, 29 of 32)
2. RMS-00-C Rev 6 chg 1 pages 79-80
3. Provide applicants with Attachments 1 and 2 of OP-2383C, "RADIATION MONITOR ALARM SETPOINT CONTROL" (Pages 28, 29 of 32)

NRC K/A System/E/A

System 073 Process Radiation Monitoring (PRM) System

Number A4.02 **RO** 3.7 **SRO** 3.7 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Radiation monitoring system control panel

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

63 ☑ RO ☑ SRO Question ID: 5000013 Origin: New ☐ Memory Level

Unit 2 had started up following a refueling outage. Given the following events and conditions:

- The upper extension shaft on CEA-01 (rod in the center of the core) was not re-coupled following the reactor refueling
- The reactor is currently operating at 100% power

Which one of the following statements correctly describes:
1. the CEA position indication for this condition, and
2. the predominant flux distribution concern (if any) if operation at 100% power continues?

A 1. The amber light will remain lit on the core mimic ☐
2. Axial flux will be suppressed at the bottom of the core

B 1. The amber light will remain lit on the core mimic ☐
2. Radial flux peaking will occur in the center of the core

C 1. CEA position indication will show that CEA-01 is fully withdrawn ☑
2. Radial flux peaking will occur at the outer core regions

D 1. CEA position indication will show that CEA-01 is fully withdrawn ☐
2. There will be no abnormal flux distribution in the core

Justification

The upper extension shaft contains the magnet that the RPIS reed switches. If the upper extension shaft is not coupled to the CEA, there will be no direct indication of the problem because the upper extension shaft will actuate the reed switches as it is withdrawn. CEA XX is in the center of the core and the flux will be locally depressed by this rod remaining inserted, there by causing a radial flux distribution problem.

CHOICE [A] - NO
WRONG The amber rod bottom light will not remain lit. If the applicant does not understand the difference between axial and radial flux suppression, the applicant may select this distracter because it is the only distracter that states axial flux is suppressed in the bottom of the core.

CHOICE [B] - NO
WRONG - The amber rod bottom light will not remain lit - CEA position indication will appear as if the rod is fully withdrawn - but the rod remains fully inserted. Radial flux suppression occurs in the center of the core causing radial flux peaking in the outer regions of the core.

CHOICE [C] - YES
CORRECT- The amber rod bottom light will not be lit and there will be no direct indication that the CEA remains fully inserted. The CEA will appear fully withdrawn on the core mimic, PPC and CEAPS displays. Radial flux will be suppressed in the center of the core - and will therefore peak in the outer regions of the core. If the candidate does not understand flux peaking and thinks that the reed switches are actuated by the CEA and not the upper extension shaft, the applicant could pick this distracter.

CHOICE [D] - NO
WRONG - This choice is plausible if the applicant thinks that uncoupling the upper extension shaft does not prevent the control rod from being withdrawn.

References

1. CED-01-C.rlc04 pages 26-28, 53, 56
2. Figures 23 and 24 M2 04-02-98 86001050/51
3. Core Mimic - Figure 23

NRC K/A System/E/A

System 014 Rod Position Indication System (RPIS)

Number A1.04 **RO** 3.5 **SRO** 3.8 **CFR Link** (CFR: 41.5/45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: Axial and radial power distribution

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

64 ☑ RO ☑ SRO Question ID: 5000065 Origin: New ☑ Memory Level

Unit 2 is in MODE 5 preparing to commence a refueling outage. Given the following events and conditions:

- A containment purge is being lined up in accordance with OP-2314B, "Containment and Enclosure Building Purge" using the containment cleanup mode
- Noble gas concentration inside containment exceeds the limits in OP-2314B

Which one of the following statements correctly describes the purge path required?

A Containment purge flow will be directed through the EBFAS system through 2-AC-3 (EB PURGE SUPPLY DMPR) ☑

B Containment purge flow will be directed through the EBFAS system through 2-AC-57 (CTMT PURGE EXH DMPR) ☐

C Containment purge flow will be directed through the main exhaust system through 2-AC-3 (EB PURGE SUPPLY DMPR) ☐

D Containment purge flow will be directed through the main exhaust system through 2-AC-57 (CTMT PURGE EXH DMPR) ☐

Justification

CHOICE [A] - CORRECT
If noble gas exceeds 1.25E2 Ci/cc, then purging through the EBFAS system is required and 2-AC-57 must be red-tagged shut.

CHOICE [B] - WRONG
Partially correct - containment purge flow is directed through the EBFAS system but 2-AC-57 must be red tagged shut. The path through the EBFAS system is correct but 2-AC-57 is not in this path.

CHOICE [C] - WRONG
Partially correct - containment purge is not directed through the main exhaust system but AC-3 is used to control the release flow rate.

CHOICE [D] - WRONG
Containment purge is not directed through the main exhaust system if noble gas concentration exceeds the limits in OP-2314B, but if the main exhaust was used, AC-57 is the correct exhaust damper. This is the normal purge release path for containment.

References

1. OP2314B, "Containment and Enclosure Building Purge", Revision 19-03 (9/17/04) (Pg 9,10 of 63)
2. RWV-00-C rev 4 page 11
3. RWV Figures 1, 2, 7

NRC K/A System/E/A

System 029 Containment Purge System (CPS)

Number A4.01 **RO** 2.5 **SRO** 2.5 **CFR Link** (CFR: 41.7 / 45.5 to 45.8)

Ability to manually operate and/or monitor in the control room: Containment purge flow rate

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

65 ☑ RO ☑ SRO Question ID: 0071625 Origin: Bank ☑ Memory Level

Unit 2 was conducting refueling and a full core reload is in progress when a design-basis earthquake occurs. Given the following events and following conditions:

- A spent fuel assembly had been lifted by the refuel machine and moved beyond the reactor vessel flange
- The upender was on the refuel pool side and empty
- The refuel machine operator noticed the level in the reactor cavity is lowering at a rate of about 5 inches/minute
- The location of the leak was reported to be from the reactor cavity pool seal
- AOP 2578 (Loss of Refuel Pool and Spent Fuel Pool Level) was entered
- The refueling SRO has directed the refueling machine operator to place the spent fuel assembly at the PRE-PROGRAMMED SAFE POINT and evacuate containment

In accordance with AOP-2578, which of the following choices correctly describes necessary actions to comply with the directions of the refueling SRO?

A Transport to the designated safe point in the CORE and ungrapple the fuel assembly. ☐

B Transport to the designated safe point in the NORTH SADDLE AREA of the refuel pool and ungrapple the fuel assembly. ☐

C Transport to the designated safe point in the SFP REGION B STORAGE RACKS and do NOT ungrapple the fuel assembly. ☐

D Transport to the designated safe point in the SOUTH SADDLE AREA of the refuel pool and do NOT ungrapple the fuel assembly. ☑

Justification

CHOICE [A] - WRONG
Returning the spent fuel assembly to the core is not the required action per AOP 2578 unless the assembly has not yet been moved beyond the reactor cavity flange. In any case, the fuel assembly should not be ungrappled. This action was the correct answer to one version of the bank question.

CHOICE [B] - WRONG
The north saddle area is not the quickest way to isolate the leak and restow the bundle. The upper guide assembly is stored in the north saddle area. The safe point is in the south saddle area. In any event, the fuel assembly should not be ungrappled.

CHOICE [C] - WRONG
With the upender on the refuel pool side, the priority is to close the refueling canal gate. Transfering the fuel assembly to the spent fuel pool would only delay this action.

CHOICE [D] - CORRECT
This is the correct action per AOP-2578.

This bank question has been used repeatedly with correct answers as A (store in the SFP) and B (store in the core). Storing the fuel assembly in the south saddle area is not one of the correct answers among the bank questions. However, the programmed "safe point" for the refueling machine is the south saddle area. This is where the fuel assembly would be stored if there was not enough time to put the assembly back in the core or move it to the spent fuel pool. The purpose of this question is to determine if the applicants understand when it would be appropriate to store the fuel assembly at the south saddle safe point.

Note: There is a precaution in AOP-2578 that states:

"3.5 A refuel cavity drain line failure in the south saddle would drain that area completely. This eliminates this area and the transfer canal as a safe storage location."

The south saddle are has only 1-2 ft of water over the fuel bundles to protect and shield them if water drops to the level of the reactor cavity flange - this is not an appropriate storage location when the location of the leak is from the south cavity drain. This selection has never been a correct answer for any prior version of this bank question.

AOP-2578 tasks the refueling SRO with the following priorities in the even to a loss of refueling water level:

NOTE
The Refueling SRO is tasked with assessing water level loss rate, time available to perform actions, and potential safe locations for reactor lower internal components not in the reactor. If worst case leak rates are present, the following priorities are provided and should be performed in parallel:
1. Ensuring Transfer Carriage is in the SFP side.

2. Closing Transfer Tube Isolation, 2-RW-280.
3. Lowering an irradiated fuel assembly in the refueling machine into the reactor vessel only if sufficient space is available to allow quick insertion of the fuel assembly (i.e., reload vs. shuffle).
4. Lowering an assembly in the refueling machine into the south saddle.
5. Lowering an assembly from the platform crane into a SFP storage location.

References

1. AOP-2578, "Loss of Refuel Pool and Spent Fuel Pool Level", Revision 6 (05/23/03) Step 4.1.9 (Pg 8,9,10 of 15)
2. REF-04-C rev 3 pages 48-50

NRC K/A System/E/A

System 034 Fuel Handling Equipment System (FHES)

Number A1.02 **RO** 2.9 **SRO** 3.7 **CFR Link** (CFR: 41.5 / 45.5)

Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Fuel Handling System controls including: Water level in the refueling canal

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

66 ☑ RO ☑ SRO Question ID: 5000069 Origin: New ☐ Memory Level

Unit 2 is conducting a reactor start up. Given the following events and conditions:

- Wide range (WR) logarithmic nuclear instrument (NI) channels C and D are out of service
- The reactor is not yet critical
- The ECP expected critical rod height is 100 steps on Regulating Group 6
- Regulating Group 4 is withdrawn to 80 steps
- WR NI Channel A failed low

WRL NI Channel A	<1.0E-1	CPS
WRL NI Channel B	6.2E2	CPS

Which one of the following statements correctly describes the required action (if any) required to comply with Technical Specifications?

A Immediately trip the reactor. ☐

B Insert all control rods and shutdown the reactor. ☐

C Stop the startup until WRL NI Channel A has been repaired. NO other actions are required. ☐

D Immediately ensure adequate shutdown margin. ☑

Justification

Tech Spec 3.3.1 requires 2 channels of WR NI's to be operable in MODEs 3, 4 and 5. The reactor does not reach MODE 2 until group 4 CEAs are withdrawn to 92 steps. Tech Spec LCO 3.3.1.1 applies under these conditions and requires immediate determination of adequate shutdown margin. There are no other Tech Spec requirements that apply to this case.

CHOICE [A] - NO
WRONG Tech specs require 2 channels of WRL NI's to be oprable in MODE 3. Tripping the reactor is not required - only determining adequate shutdown margin.

CHOICE [B] - NO
WRONG Tech specs require 2 channels of WRL NI's to be operable in mode 3. Immediately determining shutdown margin is required with the reactor in modes 3.

CHOICE [C] - NO
WRONG - Tech Spec 3.3.1.1 requires 2 WRL NI channels to be operable in mode 3. Stopping the startup may be a prudent action but it is not required by Tech Spec 3.3.1.1.

CHOICE [D] - YES
CORRECT The reactor has not yet transitioned to mode 2 and 2 WRL NI channels are required in mode 3. With only 1 WRL NI channel operable, the tech spec action is to immediately determine shutdown margin.

References

1. OP-2202 rev 20-05 pages 4, 7-8,17, Att 1, Att 4
2. NIS-01-C pages 9-16, 30-31
3. ARP C-04 AB-12

NRC K/A System/E/A

System 033

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.22 **RO** 3.4 **SRO** 4.1 **CFR Link** (CFR: 43.2 / 45.2)

Knowledge of limiting conditions for operations and safety limits

67 ☑ RO ☑ SRO Question ID: 5000070 Origin: New ☐ Memory Level

Unit 2 is preparing to discharge the contents of the CLWMT though the liquid radwaste system. Given the following conditions and events:

- The discharge tank contains 0.1 μCi/ml of Tritium as the primary isotope

Which one of the following statements correctly describes:
1. The biological hazard presented by the contents of the tank
2. The administrative controls that ensure the activity limits of 10CFR20 will NOT be exceeded?

A 1. Tritium emits beta radiation and primarily presents an internal exposure hazard ☑
2. The REMODCM limits the total amount of Tritium that can be released

B 1. Tritium emits gamma radiation and primarily presents a whole body exposure hazard ☐
2. The NPDES limits the maximum volume for every batch to 5000 gallons that can be discharged

C 1. Tritium emits beta radiation and primarily presents an internal exposure hazard ☐
2. The NPDES limits the maximum volume for every batch to 5000 gallons that can be discharged

D 1. Tritium emits gamma radiation and primarily presents a whole body exposure hazard ☐
2. The REMODCM limits the total amount of Tritium that can be released

---

Justification

CHOICE [A] - CORRECT
Tritium decays by emission of a low energy beta particle which only presents an internal exposure problem to the human body. TRM 3/4.11 limits the total activity of Tritium that can be released to ensure that the limits of 10CFR20 are not exceeded.

CHOICE [B] - WRONG
Tritium emits a low energy beta particle - not a gamma. NPDES does not limit that total amount of activity that can be discharged - only limits other biological threats to the environment like chemicals and temperature of disharge water.

CHOICE [C] - WRONG
Partially correct - Tritium does emit a low energy beta particle which presents an internal hazard. NPDES does not limit the amount of activity that can be dicharged to ensure 10CFR20 is not exceeded.

CHOICE [D] - WRONG
Partially correct - Tritium does not emit gamma radiation and does not present a problem to whole body exposure. However, Tech Spec 3.11.1 is the guiding limit to prevent exceeding the 10CFR20 limits.

References

1. RLD-04-C rev 1 pages 25-26
2. TRM 3/4.11

NRC K/A System/E/A

System 068 Liquid Radwaste System (LRS)

Number K5.04 **RO** 3.2 **SRO** 3.5 **CFR Link** (CFR: 41.5 / 45.7)

Knowledge of the operational implication of the following concepts as they apply to the Liquid Radwaste System: Biological hazards of radiation and the resulting goal of ALARA

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

68 ☑ RO ☑ SRO Question ID: 5000071 Origin: New ☐ Memory Level

Unit 2 was operating at 100% power in the process of conducting a gaseous waste discharge. Given the following events and conditions:

- The operators are releasing the contents of waste gas decay tank (WGDT) A in accordance with SP-2617B (Gaseous Waste Discharge)
- RM-9095 (Waste Gas Radiation Monitor) lost power 3 minutes after the release had started

Which one of the following statements correctly describes:
1. The complete list of automatic actions that will occur, and
2. The actions required to restart the release?

A 1. 2-GR-37.1, 2-GR-37.2 (waste gas discharge isolation valves) AND 2-GR-8.1A (waste gas decay tank's outlet valve) will close.
2. The waste gas discharge may be restarted provided 2 independent samples of the WGDT have been taken and analyzed. ☑

B 1. 2-GR-37.1, 2-GR-37.2 (waste gas discharge isolation valves) AND 2-GR-8.1A (waste gas decay tank's outlet valve) will close.
2. The waste gas discharge may not be restarted until RM-9095 has been restored to an operable status. ☐

C 1. ONLY 2-GR-37.1 AND 2-GR-37.2 (waste gas discharge isolation valves) will close.
2. The waste gas discharge may be restarted provided 2 independent samples of the WGDT have been taken and analyzed. ☐

D 1. ONLY 2-GR-37.1 AND 2-GR-37.2 (waste gas discharge isolation valves) will close.
2. The waste gas discharge may not be restarted until RM-9095 has been restored to an operable status. ☐

Justification

CHOICE [A] CORRECT
The waste gas discharge isolation valves and the isolation vale to the WGDT being released will automatically close.
RM-9095 is not required to be operable to conduct a waste gas release - but 2 independent sample of the WGDT contents must be taken and verified before the release can be restarted.

CHOICE [B] - WRONG
RM-9095 is not required to be operable to conduct a waste gas release.

CHOICE [C] - WRONG
2-GR-8.1A also closes automatically in additon to 2-GR-37.1 & 2. Partially correct - the release may be restarted without RM-9095 being operable provided the WGDT is independently sampled by 2 people.

CHOICE [D] - WRONG
2-GR-8.1A also closes automatically in additon to 2-GR-37.1 & 2. RM-9095 is not required to be operable to conduct a waste gas release - but 2 independent sample of the WGDT contents must be taken and verified before the release can be restarted.

References

1. SP-2617B rev 11-08 - page 3
2. RMS-00-C rev 6 pges 40-41
3. GRW-04-C Rev 5/3 pages 13-14, 17

NRC K/A System/E/A

System 071 Waste Gas Disposal System (WGDS)

Number A2.05 **RO** 2.5* **SRO** 2.6 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

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 and PRM Systems

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

69

☑ RO ☑ SRO Question ID: 0056618 Origin: Mod ☐ Memory Level

Unit 2 was operating at 100% power. Given the following events and conditions:

- A reactor trip occurs at 0200 and the operators enter EOP-2525, "Standard Post Trip Actions"
- At 0210, RM-4299 C (main steamline radiation monitor) slowly increases from 1.0E-2 to 1.5E0 R/hr

Which one of the following statements correctly describes the cause of this trend?

A A SGTR has occurred in the "B" S/G at 0210 ☐

B A SGTR occurred on the "B" S/G prior to the trip at 0200 ☐

C A large crud burst has occurred in the RCS at 0200 ☐

D Severe fuel damage has occurred at 0210 ☑

Justification

CHOICE [A] - WRONG
The main steamline radiation monitors respond primarily to N16 gamma which decays away after the reactor trip. The N16 source term has decayed away 10 minutes after the trip. The applicant could pick this answer if he/she did not recognize that the main steam line rad monitors respond priimarily to high energy gamma radiation. RCS coolant would migrate into a ruptured S/G through the broken tube but only the gaseous activity would be transported down the main steam lines. Typical RCS gaseous activites would not be sufficient to cause the main steam line RAD monitor to alarm.

CHOICE [B] - WRONG
Applicants are trained to use the MSL RAD monitors to diagnose a SGTR - so some may recall that the MSL monitors respond to N16 if the SGTR occurs before the trip. They would be incorrect for this condition because the N16 source term has decayed away after 10 minutes and the RCS gaseous activity levels in the "A" S/G are insufficent to cause RM-4299C to respond.

CHOICE [C] - WRONG
A large crud bust does not contain volatile radionuclides and the activity from the crud burst will not migrate into the main steam lines. Particlate activity in the S/G will remain in the S/G and not enter the main steam lines. Some applicants may not recognize that only the gaseous activity enters the steam lines.

CHOICE [D] - CORRECT
RM-4299C will also respond to shine from Containment (streaming through steam line penetrations), during severe accidents with fuel damage. RM-4299C is used as an alternate Containment high range monitor because of this response.

References

1. RMS-00-C rev 6 page 56-57

NRC K/A System/E/A

System 072 Area Radiation Monitoring (ARM) System

Number K1.05 **RO** 2.8* **SRO** 2.9* **CFR Link** (CFR: 41.2 to 41.9 / 45.7 to 45.8)

Knowledge of the physical connections and/or cause- effect relationships between the ARM system and the following systems: MRSS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

70 ☑ RO ☑ SRO Question ID: 0054466 Origin: Mod ☑ Memory Level

The unit is operating at 100% power with all circulating water pumps in operation when the Main Condenser Waterbox 'D' Circ Water Inlet Valve, CW-11E, red position indicating light on Panel C-06 is noted out. Shortly thereafter, the following alarms are received:

- WEST COND PIT SUMP LVL HI (C-06/7, BB-22)
- CIRC WATER PUMP A OVERLOAD/TRIP (C-06/7, A-9)
- CIRC WATER PUMP B OVERLOAD/TRIP (C-06/7, B-9)
- CIRC WATER PUMP C OVERLOAD/TRIP (C-06/7, C-9)
- CIRC WATER PUMP D OVERLOAD/TRIP (C-06/7, D-9)

Which of the following explains the cause of the trip of the circulating water pumps and the primary reason for this protection?

A switches actuated by level at 6 inches below the top of the west cond pit sump, to protect AFW pumps ☐

B switches actuated by level at 10 inches in the cond pit area, to protect AFW pumps ☑

C switches actuated by level at 6 inches below the top of the west cond pit sump, to protect main generator auxiliary systems ☐

D switches actuated by level at 10 inches in the cond pit area, to protect main generator auxiliary systems ☐

Justification

ARP 2590E states any 2 of 4 level switches reaches 10", all CW pumps will trip.

CHOICE (A) - NO
WRONG: Level must reach 10 inches in the condenser pit area to trip all circulating water pumps.
VALID DISTRACTOR: Level at 6 inches below the top of the west cond pit sump actuates a level switch to cause the sump level high alarm.

CHOICE (B) - YES
2 of 4 level switches in the condenser pit area actuated at >10 inches in pit will trip all circulating water pumps to prevent flooding from challenging the continued operability of the AFW pumps.

CHOICE (C) - NO
WRONG: The primary reason is to protect AFW pumps, not main generator auxiliaries.
VALID DISTRACTOR: Level at 6 inches below the top of the west cond pit sump actuates a level switch to cause the sump level high alarm.

CHOICE (D) - NO
WRONG: Level must reach 10 inches in the condenser pit area to trip all circulating water pumps.
VALID DISTRACTOR: Applicant may think that the purpose is to protect main generator auxiliary systems since the AFW pumps are within protected rooms.

References

1. CWS-00-C, "Circulating Water and Water Box Priming Systems, Revision 9 (8/29/01), Section D.3.b. "Response to High Waer Level in Condenser Pit Area (Pg 29 of 45)
2. ARP-2590E-116, "WEST COND PIT SUMP LVL HI"

NRC K/A System/E/A

System 075 Circulating Water System

Number K3.07 **RO** 3.4* **SRO** 3.5* **CFR Link** (CFR: 41.7 / 45.6)

Knowledge of the effect that a loss or malfunctions of the circulating water system will have on the following: ESFAS

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

71 ☑ RO ☑ SRO Question ID: 5000067 Origin: New ☑ Memory Level

A transfer of a new fuel assembly is in progress from one location in the spent fuel pool to another using OP-2303B, "SFP Fuel Handling Operations". The operator raises the hoist with the desired assembly grappled until upward motion is stopped by the upper limit switch interlock.

What must be done next?

A Release hoist raise switch, use the bridge/trolley controls to move to destination. ☐

B Stop all hoist and crane movement and notify Reactor Engineering immediately. ☑

C Lower assembly into initial location and contact Reactor Engineering for resolution. ☐

D Slowly lower hoist until load cell indicates 250 to 290 pounds, then continue move. ☐

Justification

CHOICE (A) - NO
WRONG: Procedure directs stopping all fuel movement.
VALID DISTRACTOR: Plausible that it is acceptable to have motion stopped by interlock.

CHOICE (B) - YES
Danger of ungrappling and dropping fuel assembly. Must stop and notify immediately.

CHOICE (C) - NO
WRONG: Procedure directs stopping all fuel movement.
VALID DISTRACTOR: Plausible that corrective action would be to lower into rack. This is correct action for fuel handling event.

CHOICE (D) - NO
WRONG: Procedure directs stopping all fuel movement.
VALID DISTRACTOR: An applicant may think that use of the interlock affects the load cell. 250 to 290 pounds is the load identified by the procedure for a suspended assembly.

References

1. OP-2303B, "SFP Fuel Handling Operations", Revision 1 (10/18/03) (Pg 10,11 of 35)
2. REF-04-C, "Refueling Equipment" Lesson, Revision 3 (Pg 10 of 71)

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.28 **RO** 2.6 **SRO** 3.5 **CFR Link** (CFR: 43.7 / 45.13)

Knowledge of new and spent fuel movement procedures.

72 ☑ RO ☑ SRO Question ID: 5000066 Origin: New ☐ Memory Level

Refueling is in progress. A new fuel assembly has just been lowered into core location A-11 (core map attached). You are the PPO and have noted the following before and after readings on the wide range logarithmic power channels:

	BEFORE	AFTER
WR CH A	1.9E1 cps	2.0E1 cps
WR CH B	1.8E1 cps	3.2E1 cps
WR CH C	1.6E1 cps	1.9E1 cps
WR CH D	1.0E1 cps	1.2E1 cps

Based on these indications, which of the following is required?

A Suspend all core alterations and positive reactivity additions. ☐

B Commence boration per AOP-2558, "Emergency Boration". ☐

C Continue to monitor nuclear instruments, NO immediate action required. ☑

D Withdraw the fuel assembly and contact Reactor Engineering for guidance. ☐

Justification

CHOICE (A) - NO
WRONG: Counts have not doubled. The only appreciable increase in counts is on CH B which is immediately adjacent to the location of the new assembly.
VALID DISTRACTOR: Per OP-2209A, if at any time, unanticipated count rate multiplication, (i.e., doubling), is indicated, then suspend refuel operations.

CHOICE (B) - NO
WRONG: Counts have not doubled. The only appreciable increase in counts is on CH B which is immediately adjacent to the location of the new assembly.
VALID DISTRACTOR: Per OP-2209A, if at any time, unanticipated count rate multiplication, (i.e., doubling), is indicated, then commence boration

CHOICE (C) - YES
Criteria for action is observation of an unanticipated count rate multiplication, (i.e., doubling). Counts have not doubled.

CHOICE (D) - NO
WRONG: Counts have not doubled. The only appreciable increase in counts is on CH B which is immediately adjacent to the location of the new assembly.
VALID DISTRACTOR: Plausible that requirement is to remove the assembly to lower core reactivity while situation is evaluated.

References

1. OP-2209A, "Refueling Operations", Revision 24 (12/17/03) (Pg 25 of 64)
2. Provide copy of NIS-01-C Lesson Figure 2 to applicant.

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.30 **RO** 3.5 **SRO** 3.3 **CFR Link** (CFR: 45.12)

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

73 ☑ RO ☑ SRO Question ID: 5000064 Origin: New ☑ Memory Level

Which of the following conditions would NOT require immediate entry into EOP-2525, "Standard Post-Trip Actions", if the condition were to occur inadvertently with the reactor operating at 100% power?

A Containment Isolation Actuation Signal on both Facilities ☑

B Main Steam Isolation Signal on Facility 1 only ☐

C Overcurrent trip of normal feeder breaker to Bus 25B from NSST ☐

D Loss of VA-20 with loss of HV to Linear Range CH 'D' ☐

Justification

CHOICE (A) - YES
Spurious CIAS is addressed by AOP-2571, "Inadvertent Emergency Core Cooling System Initiation", which provides direction for maintaining power operation while addressing the problems of inadvertent isolation.

CHOICE (B) - NO
WRONG: Each facility of ESAS can complete the safety function. Both MSIVs will close resulting in either a manual or automatic reactor trip.
VALID DISTRACTOR: Applicant may think that both Facilities must actuate to trip reactor.

CHOICE (C) - NO
WRONG: An overcurrent trip on the bus normal feed will lock out the alternate feed from the RSST. Fast transfer will not occur. The bus will de-energize and its associated RCP breakers will trip. The reactor will trip on loss of low speed of two RCPs.
VALID DISTRACTOR: Applicant may think that fast tranfer will occur, maintaining power to the RCPs.

CHOICE (D) - NO
WRONG: Loss of VA-20 trips RPS CH 'B'. Loss of HV to a linear range NI causes associated channel trips through the PTTI. With 2 RPS high power trips, a reactor trip will occur.
VALID DISTRACTOR: Applicant may think that loss of HV will not trip the variable high power on Channel D since the NI output will fail low.

References

1. AOP-2571, "Inadvertent Emergency Core Cooling System Initiation", Revision 4 (12/14/98) (Pg 4 of 15)
2. IHE-00-C, "In-House Electrical System" Lesson, Revision 9 (Pg 10 of 86)
3. Source: INPO Bank - Q# 16 - Used at Braidwood 1, 6/7/1999

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.2 **RO** 3.9 **SRO** 4.1 **CFR Link** (CFR: 41.7 / 45.7 / 45.8)

"Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. Note: The issue of setpoints and automatic safety features is not specifically covered in the systems sections)."

74 ☑ RO ☑ SRO Question ID: 5000063 Origin: New ☑ Memory Level

The unit is operating at power when multiple indications are received (radiation levels, feed system response) of a SGTR in #2 SG. The US orders a manual reactor trip and SIAS. The crew enters EOP-2525, "Standard Post Trip Actions".

During initial performance of EOP-2525 Step 3 ("Determine Status of RCS Inventory Control") numerous additional annunciators alarm. Operators observe a rapid depressurization of #1 SG with a concurrent rapid rise in containment pressure.

Based on these conditions, the crew will immediately ________________.

A return to Step 1 of EOP-2525 and commence the procedure again ☑

B return to Step 1 of EOP-2525 but bring a step forward for manual CIAS ☐

C transition to an ORP based on Appendix 1, "Diagnostic Flowchart" ☐

D transition to an FRP based on Appendix 1, "Diagnostic Flowchart" ☐

Justification

CHOICE (A) - YES
OP-2260, Step 1.19.1 directs the action.

CHOICE (B) - NO
WRONG: EOP-2525 does not contain any "CONTINUOUSLY APPLICABLE" steps.
VALID DISTRACTOR: When in ORP or FRP, selected steps are identified as "CONTINUOUSLY APPLICABLE".

CHOICE (C) - NO
WRONG: OP-2260 directs performance of EOP-2525. Final steps then direct use of diagnostic flowchart.
VALID DISTRACTOR: For some events, transition to an ORP is the course of action that is concluded using the diagnostic flowchart.

CHOICE (D) - NO
WRONG: Appendix 1 is not used until completion of SPTAs.
VALID DISTRACTOR: Plausible that an FRP will be indicated to deal with multiple events in progress.

References

1. OP-2260, "Unit 2 EOP User's Guide", Revision 8 (7/11/02) (Pg 16,18 of 32)

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.12 **RO** 3.4 **SRO** 3.9 **CFR Link** (CFR: 41.10 / 45.12)

Knowledge of general operating crew responsibilities during emergency operations.

 # 75 ☑ RO ☑ SRO Question ID: 5000062 Origin: New ☐ Memory Level

The plant was manually tripped from 100% power due to a tube rupture in #1 SG estimated to be approximately 200 gpm at the time of the trip. Upon entry into EOP 2534, "Steam Generator Tube Rupture," the following conditions exist:

- All electrical buses are energized.
- B and D RCPs are operating.
- #1 SG level is 46% and stable.
- #2 SG level is 52% and stable.
- SG pressures are 895 psia, steady.
- Pressurizer level is off scale low.
- Pressurizer pressure is 1550 psia, slowly lowering.
- SIAS, CIAS, and EBFAS have fully actuated.
- Th is 534° F and stable.
- Tc is 532°F and stable.
- Containment pressure is 0.1 psig and stable.
- Containment temperature is 95 °F and stable.
- Containment radiation monitors are NOT in alarm and NOT rising.
- Steam Jet Air Ejector and Blowdown radiation monitors are in alarm.
- Radiation monitors outside Containment are NOT in alarm, NOT rising.

Action steps in EOP 2534 appear as follows:

*6. RCP Trip Strategy
*7. Align Condenser Air Removal to Unit 2 Stack
8. Commence cooldown to Th less than 515°F
*9. Reduce and control RCS pressure

Which of the following actions is the correct sequencing to ensure optimal control and mitigation of this event?

A ☐
1. Commence an RCS cooldown at the maximum controllable rate.
2. Depressurize the RCS using Main Spray.
3. Ensure at least one RCP in each loop is operating.
4. Open Condenser Air Removal to Unit 2 stack, EB-57 and start at least one Main Exhaust Fan.

B ☐
1. Ensure at least one RCP in each loop is operating.
2. Commence an RCS cooldown at the maximum controllable rate.
3. Depressurize the RCS using Main Spray.
4. Open Condenser Air Removal to Unit 2 stack, EB-57 and start at least one Main Exhaust Fan.

C ☐
1. Commence an RCS cooldown at the maximum controllable rate.
2. Ensure at least one RCP in each loop is operating.
3. Open Condenser Air Removal to Unit 2 stack, EB-57 and start at least one Main Exhaust Fan.
4. Depressurize the RCS using Main Spray.

D ☑
1. Ensure at least one RCP in each loop is operating.
2. Open Condenser Air Removal to Unit 2 stack, EB-57 and start at least one Main Exhaust Fan.
3. Commence an RCS cooldown at the maximum controllable rate.
4. Depressurize the RCS using Main Spray

Justification

A is incorrect. The examinee may believe that it is necessary to commence a cooldown and to depressurize the RCS to preserve a safety function. The step to commence a cooldown is NOT asterisked; therefore, it must be performed in sequence.

B is incorrect. The examinee may believe that the given sequence will preserve a safety function.

C is incorrect. The examinee may believe that it is necessary to commence a cooldown to preserve a safety function. This step is NOT asterisked; therefore, it CANNOT be performed out of sequence.

D is correct. Per OP 2260, Unit 2 EOP User Guide, states, "Procedural steps listed in alphanumeric order are sequential steps and shall be addressed in that sequence. Exceptions to this are as follows: 1) Steps which are asterisked may be brought forward to correct or preserve a safety function. 2) Steps may be performed out of order after they have been accomplished once, if they are continuously applicable steps as indicated by an asterisk." In this case, none of the steps listed need to be performed to preserve a safety function. Additionally, none of the listed steps needs to be performed out of order as none have been performed yet.

References

1. EOP-2534, "Steam Generator Tube Rupture", Revision 9 (2/27/01) (Pg 3, 20, 21 of 37)
2. OP-2260, "Unit 2 EOP User's Guide", Revision 8 (7/11/02) (Page 10,11 of 32)

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.19 **RO** 2.7 **SRO** 3.7 **CFR Link** (CFR: 41.10 / 45.13)

"Knowledge of EOP layout, symbols, and icons."

 # 76 ☐ RO ☑ SRO Question ID: 0055020 Origin: Mod ☐ Memory Level

A reactor shutdown is in progress with the following conditions:

- Rx is at 0.8% power.
- Rods are being inserted in Manual Sequential mode.
- Group 7 CEAs are at 30 steps.
- Group 6 CEAs are at 155 steps.
- A Group 6 CEA drops to the bottom of the core.

The PPO releases the CEA Control Switch, depresses the Regulating Group 6 INHIBIT BYPASS button, then places and holds the CEA Control Switch in INSERT.

Which one of the following describes the CEDS response and Tech Spec implications?

A ONLY Group 6 CEAs will insert, restore CEA Motion Inhibit to operable within 6 hours or ensure the Unit in MODE 3 within next 6 hours ☐

B ONLY Group 7 CEAs will insert, restore CEA Motion Inhibit to operable within 6 hours or ensure the Unit in MODE 3 within next 6 hours ☐

C BOTH Group 6 and Group 7 CEAs will insert, ensure Group 6 CEAS restored to within 10 steps of each other within 2 hours or ensure the Unit in MODE 3 within next 6 hours ☐

D NEITHER Group 6 nor Group 7 CEAs will insert, ensure Group 6 CEAS restored to within 10 steps of each other within 2 hours or ensure the Unit in MODE 3 within next 6 hours ☑

Justification

SRO ONLY QUESTION - Samples 55.43(2), Facility operating limitations in the technical specifications and their bases.

CHOICE (A) - NO
WRONG: CMI bypass requires the operation of both the individual group Inhibit Bypass and the overall system CMI bypass buttons.
VALID DISTRACTOR: Applicant may think that PPO actions bypassed CMI for Group 6. Applicant may think actions required for CMI operability.

CHOICE (B) - NO
WRONG: CMI bypass requires the operation of both the individual group Inhibit Bypass and the overall system CMI bypass buttons.
VALID DISTRACTOR: Applicant may think only Group 6 affected by this CMI. Applicant may think actions required for CMI operability.

CHOICE (C) - NO
WRONG: CMI bypass requires the operation of both the individual group Inhibit Bypass and the overall system CMI bypass buttons.
VALID DISTRACTOR: Applicant may think CMI is bypassed for Group 6 and CMI is not active for Group 7.

CHOICE (D) - YES
CMI bypass requires the operation of both the individual goup Inhibit Bypass and the overall system CMI bypass buttons. PPO actions did not bypass CMI for any group. TS requires restoring CEA to within proper alignment within 2 hours or taking unit to MODE 3 within following 6 hours.

References

1. CED-01-C, "Control Element Drive System" Lesson, Revision 4 (1/26/04) (Pg 28, 30, 55, 56, 57 of 67)
2. TS 3.1.3.1, "Movable Control Assemblies" - "CEA Position"

NRC K/A System/E/A

System 003 Dropped Control Rod

Number AA2.02 **RO** 2.7 **SRO** 2.8 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Dropped Control Rod: Signal inputs to rod control system

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

77 ☐ RO ☑ SRO Question ID: 5000017 Origin: New ☐ Memory Level

The TS limiting condition for operation action requirements for a Safety Injection Tank (SIT) are less restrictive on time allowed to restore to OPERABLE status for boron concentration than for low level. This is because the ________________

A tank volume is expected to immediately flush through core and out the break where it will mix with the rest of the RCS inventory. ☐

B tank volume requirements are based on one tank emptying through the break and a passive failure of a second tank. ☐

C boron requirements consider the average concentration in the total volume of three safety injection tanks. ☑

D boron requirements assume sufficient shutdown margin due to void fraction during a large break LOCA. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(2), Facility operating limitations in the technical specifications and their bases.

CHOICE (A) - NO
WRONG: Boron concentration is a requirement to ensure ability to maintain subcriticality during a LOCA. However, the effects of reduced concentration during core reflood are minor for two reasons. Core boiling tends to concentrate the boron in the inventory in the vessel and the boron concentration limit is based on the average concentration in the total volume of three safety injection tanks.
VALID DISTRACTOR: because the applicant may attribute the reduced emphasis on concentration to the short time that the tank inventory will remain in the vessel.

CHOICE (B) - NO
WRONG: Tank volume requirements assume 3 tanks remain available to provide core cooling during the initial stages of a large break LOCA.
VALID DISTRACTOR: because the applicant may remember that design assumptions are made regarding loss of one leg of ECCS injection and on passive and active failures.

CHOICE (C) - YES
Boron concentration is a requirement to ensure ability to maintain subcriticality during a LOCA. However, the effects of reduced concentration during core reflood are minor for two reasons. Core boiling tends to concentrate the boron in the inventory in the vessel and the boron concentration limit is based on the average concentration in the total volume of three safety injection tanks.

CHOICE (D) - NO
WRONG: Boron concentration is a requirement to ensure ability to maintain subcriticality during a LOCA. However, the effects of reduced concentration during core reflood are minor for two reasons. Core boiling tends to concentrate the boron in the inventory in the vessel and the boron concentration limit is based on the average concentration in the total volume of three safety injection tanks.
VALID DISTRACTOR: because the applicant may assume that the basis for the boron requirements takes void fraction into account.

References

1. Technical Specifications Section B3/4.5.1, "Safety Injection Tanks", Amendment 220 (Page B3/4-5-1)

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.25 **RO** 2.5 **SRO** 3.7 **CFR Link** (CFR: 43.2)

Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.

78

☐ RO ☑ SRO Question ID: 5000020 Origin: Mod ☐ Memory Level

The plant is at 100% with Containment Pressure Channel "A" (PT-8113) failed high. The channel's associated bistable red "TRIP" light is lit at the Channel A ESF sensor cabinet. The pressure bistable key bypass switch associated with this channel is in INHIBIT.

The crew receives the SIAS OR UV ACTUATION SIG CH 2 TRIP annunciator on C-01. The following are the pressures sensed by the Containment Pressure channels:

PT-8113, Containment Pressure CH A => OOS
PT-8114, Containment Pressure CH B => 1.1 psig
PT-8115, Containment Pressure CH C => 1.1 psig
PT-8116, Containment Pressure CH D => 4.3 psig
PT-8117, Containment Pressure LR => 1.2 psig
PT-8238, Containment Wide Range Pressure => 1.0 psig
PT-8239, Containment Wide Range Pressure => 1.1 psig

Containment radiation monitors are NOT in alarm and stable. Containment sump level is 46% and steady. NO ECCS equipment is actuated.

Which of the following statements is correct about the "SIAS OR UV ACTUATION SIG CH 2 TRIP" annunciator on C-01 and what procedural actions are necessary to address plant conditions?

A The alarm is valid, 1 of the associated Containment Pressure Channels are tripped, enter OP-2314B, "Containment and Enclosure Building Purge". ☐

B The alarm is NOT valid, 1 of the associated Containment Pressure Channels are tripped, enter OP-2314B, "Containment and Enclosure Building Purge". ☑

C The alarm is valid, 1 of the associated Containment Pressure Channels are tripped, enter EOP-2525, "Standard Post Trip Actions". ☐

D The alarm is NOT valid, 1 of the associated Containment Pressure Channels are tripped, enter EOP-2525, "Standard Post Trip Actions". ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5), Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Alarm actuated by 2 of 4 ESF channels (8113 thru 8116) equal to or greater than 3.8 psig. However, PT-8116 is the only ESF channel above the SIAS actuation setpoint. The alarm is not valid.
VALID DISTRACTOR: One channel is reading above 3.8 psig. A rapid downpower is required because of high energy line break in containment.

CHOICE (B) - YES
Alarm actuated by 2 of 4 ESF channels (8113 thru 8116) equal to or greater than 3.8 psig. However, PT-8116 is the only ESF channel above the SIAS actuation setpoint. With less than 2 safety channels above setpoint, the alarm is not valid. A rapid downpower is required because of high energy line break in containment.

CHOICE (C) - NO
WRONG: Alarm actuated by 2 of 4 ESF channels (8113 thru 8116) equal to or greater than 3.8 psig. However, PT-8116 is the only ESF channel above the SIAS actuation setpoint. With less than 2 safety channels above setpoint, the alarm is not valid.
VALID DISTRACTOR: ESF Pressure Transmitter 8116 is reading above 3.8 psig.

CHOICE (D) - NO
WRONG: A high energy line break in containment with pressure at 1.0 psig is reason to perform rapid downpower. The plant is still at power. No reason to enter EOP-2525.
VALID DISTRACTOR: Alarm actuated by 2 of 4 ESF channels (8113 thru 8116) equal to or greater than 3.8 psig. PT-8116 is the only ESF channel above the SIAS actuation setpoint. The alarm is not valid.

References

1. ARP-2590A, "Alarm Response for Control Room Panel, C-01", Window B-34, "SIAS OR UV ACTUATION SIG CH 2 TRIP (RED WINDOW)"
2. Containment Ventilation Print 25203-26028-1, Sheet 1 of 6 (J-4, J-12)
3. MP-16-CAP-SAP01, "Condition Report Initiation", Revision 1 (8/31/04)
4. MP-16-MMM, "Corrective Action"
5. OP-2384, "ESAS Operation", Revision 14 (11/13/03) (Pg 20, 21 of 30)
6. Source: Indian Point 3 NRC Exam, 12/2003

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.46 **RO** 3.5 **SRO** 3.6 **CFR Link** (CFR: 43.5 / 45.3 / 45.12)

Ability to verify that the alarms are consistent with the plant conditions.

79 ☐ RO ☑ SRO Question ID: 5000021 Origin: Bank ☐ Memory Level

The following plant conditions exist:

The plant is at 100% power. The 'B' Emergency Diesel Generator (EDG) was declared INOPERABLE yesterday at 0600. At 0800 today, the Shift Manager discovers that the conditional 24 hour surveillance operability run on the 'A' EDG, required by the 'B' EDG action statement has not been performed.

What action is required?

A The operability surveillance of A' EDG must be performed successfully by 1000 today or be in at least HOT STANDBY within the next 6 hours. ☑

B The operability surveillance of A' EDG must be performed successfully by 0900 today or be in at least HOT STANDBY within the next 6 hours. ☐

C The operability surveillance of A' EDG must be performed successfully by 1200 today or be in at least HOT STANDBY within the next 6 hours. ☐

D The operability surveillance of A' EDG must be performed successfully by 0800 tomorrow or be in at least HOT STANDBY within the next 6 hours. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(2), Facility operating limitations in the technical specifications and their bases.

CHOICE (A) - YES
'A' EDG becomes INOPERABLE at 0600 today if it is not run successfully. Action statement (e.2) reads: "Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours and.." Two hours starts from time of declared inoperability (0800 + 2 hrs = 1000).

CHOICE (B) - NO
WRONG: 'A' EDG becomes INOPERABLE at 0600 today if it is not run successfully. Action statement (e.2) reads: "Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours and.."
VALID DISTRACTOR: TS 3.0.3 requires action taken within 1 hour to achieve HOT STANDBY within the following 6 hours if TS and associated action requirement is not met.

CHOICE (C) - NO
WRONG: 'A' EDG becomes INOPERABLE at 0600 today if it is not run successfully. Action statement (e.2) for two inoperable diesels reads: "Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours and.."
VALID DISTRACTOR: Action e.2 does require HOT STANDBY with the next 6 hours. Applicant may think that 4 hours are available to perform required testing.

CHOICE (D) - NO
WRONG: 'A' EDG becomes INOPERABLE at 0600 today if it is not run successfully. Action statement (e.2) reads: "Restore one of the inoperable diesel generators to OPERABLE status within 2 hours or be in HOT STANDBY within the next 6 hours and.."
VALID DISTRACTOR: TS 4.0.3 allows 24 hours before applying required actions for a missed surveillance. However, this is a conditional surveillance in order to maintain compliance with the action. TS 4.0.3 does not apply to this situation.

References

1. Technical Specification 3/4.8.1, Pages 3/4.8-1a (Amendment 261), 3/4.8-2a (Amendment 277)
2. Source: INPO Bank - Q# 24702 - Used at Seabrook, 05/30/2003
3. Provide TS 3.8.1.1 to applicants.

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.10 **RO** 2.7 **SRO** 3.9 **CFR Link** (CFR: 43.1 / 45.13)

Knowledge of conditions and limitations in the facility license.

 # 80 ☐ RO ☑ SRO Question ID: 5000022 Origin: Bank ☐ Memory Level

The following conditions exist for a job performed on a system:

- The general area radiation levels are 10 mrem/hr
- The hot spot in the room is a pipe elbow that has radiation levels of 100 mrem/hr
- The job will be performed near the hot spot area

Assuming the time to get to and from the job site is the same for each case and all shielding placement is done at 100 mrem/hr, which ONE (1) of the following results in the LEAST amount of personnel exposure?

A The job is performed by 2 operators for 3 hours each on the job at the hot spot. ☐

B The job is performed by 2 operators for 2 hours each on the job at the hot spot and a 3rd operator reading instructions in the general room area for 2 hours. ☐

C The job is performed by 3 operators for 1 hour each on the job at the hot spot and a 4th operator reading instructions in the general room area for 1 hour. ☑

D 2 Health Physics technicians require 1.5 hours to install and remove 1 tenth thickness of lead shielding on the hot spot. The job is performed with the shielding in place by 2 operators for 3 hours each. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

CHOICE (A) - NO
'WRONG: Total dose for this plan equals 600 mrem. The lowest dose of any choice provided is 310 mrem.
VALID DISTRACTOR: This choice involves the fewest number of personnel.

CHOICE (B) - NO
WRONG: Total dose for this plan equals 420 mrem. The lowest dose of any choice provided is 310 mrem.
VALID DISTRACTOR: This choice requires less time to complete the job than 2 other choices.

CHOICE (C) - YES
This choice results in the lowest total dose of 310 mrem.

CHOICE (D) - NO
WRONG: Total dose for this plan equals 360 mrem. The lowest dose of any choice provided is 310 mrem.
VALID DISTRACTOR: This choice installs shielding to reduce the dose to workers.

References

1. Source: Indian Point 3 NRC Exam, 12/2003

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.10 **RO** 2.9 **SRO** 3.3 **CFR Link** (CFR: 43.4 / 45.10)

Ability to perform procedures to reduce excessive levels of radiation and guard against personnel exposure.

81

☐ RO ☑ SRO Question ID: 5000023 Origin: New ☑ Memory Level

Which of the following evolutions raises an immediate ALARA concern requiring notification of Health Physics Department? Consider the effects of the described action only.

A increasing CVCS letdown flow during normal power operations ☑

B starting of the HPSI pumps by SIAS during a large break LOCA ☐

C increasing SFP cooling flow during spent fuel pool fuel moves ☐

D shifting from 'C' to 'A' Charging Pump running at 75% power ☐

Justification

SRO ONLY QUESTION - Samples 55.43(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

CHOICE (A) - YES
Step 4.4.10 requires notification of Health Physics Department of any change in charging or letdown flow.

CHOICE (B) - NO
WRONG: HPSI pumps are aligned for injection from the RWST and will not affect local dose rates until post-SRAS.
EOP does not require HP notification at start of LOCA
VALID DISTRACTOR: A LOCA has the potential to raise local dose rates.

CHOICE (C) - NO
'WRONG: Notification of HP is not required for increasing SFP cooling flow.
VALID DISTRACTOR: Plausible that increasing SFP cooling flow might create ALARA concerns.

CHOICE (D) - NO
WRONG: Shifting charging pumps does not change charging flowrate and therefore does not present ALARA concerns. Both of these pumps are located in the same general area.
VALID DISTRACTOR: A caution states that HP should be notified for changing charging flow conditions.

References

1. RPM-1.1.2, "Radiation Protection Program and ALARA Program", Revision 3 (8/19/04) (Pg 5,8,9,10,11,16 of 33)
2. OP-2304E, "Charging Pumps", Revision 15 (03/09/04), Step 4.4.10 (Pg 21 of 25)

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.3 Radiation Control

Number 2.3.2 **RO** 2.5 **SRO** 2.9 **CFR Link** (CFR: 41.12 / 43.4. 45.9 / 45.10)

Knowledge of facility ALARA program.

82 ☐ RO ☑ SRO Question ID: 5000024 Origin: Bank ☐ Memory Level

Unit 2 is operating near beginning of cycle at a burnup of 600 MWD/MTU.

The following conditions exist AFTER a transient from 90% power:

- steam generator pressure is lower
- main generator megawatt output is lower
- indicated feedwater temperature is lower
- reactor coolant hot leg temperature is lower

Which one of the following events caused this plant response and what is the applicable procedure for addressing the problem? Assume NO operator action.

A condenser backpressure rise (degraded vacuum), address with ARP-2590E (A-37), "COND VACUUM LO" ☐

B sensor input to throttle pressure limiter failed (0 psig), address with ARP-2590D (DA-22), "10% TURBINE LOAD DECREASE" ☐

C feedwater heater extraction steam isolation valve closed (heater 1B), address with ARP-2590D (AA-18), "HEATER 1A LEVEL HI" ☐

D atmospheric dump stuck in intermediate position (30% open), address with ARP-2590D (B-6), "ATMOSPHERIC DUMP VALVE NOT CLOSED" ☑

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Degraded vacuum with no movement of control valves or control rods will result in no observable change in steam generator pressure. Steam flow will remain constant. Efficiency of the turbine will decrease. Turbine will perform less work. The additional energy rejected to the condenser will be removed by circulating water system. Feedwater temperature will be unchanged and reactor power will be unchanged.
VALID DISTRACTOR: Increasing backpressure will cause main generator output to decrease.

CHOICE (B) - NO
WRONG: Throttle pressure limiter is maintained in OFF during power operations to prevent undesirable load transients.
VALID DISTRACTOR: If on, the throttle pressure limiter would act to reduce turbine load.

CHOICE (C) - NO
'WRONG: Main turbine output will increase slightly with the isolation of an extraction line as extraction steam is redirected through subsequent turbine stages.
VALID DISTRACTOR: Loss of extraction will result in lower feedwater temperature.

CHOICE (D) - YES
Fully open ARV passes steam flow equivalent to approximately 7.5% reactor power. Steam flow will increase. Steam pressure will drop. With lower steam pressure, the main turbine output will drop. Feed flow will increase to maintain steam generator level. Increased feed flow with same extraction heating steam flow will result in lower feedwater temperature. The increased total steam flow will reduce average coolant temperature. The moderator temperature coefficient of reactivity will raise reactor power until equilibrium conditions are re-established. Reactor power and core delta-T will be higher, but Tave, Th and Tc will be lower.

References

1. MSS-00-C, "Main Steam System" Lesson, Revision 6, Section 19.b (Pg 27)
2. OP-2204, "Load Changes", Revision 19 (6/29/04), Attachment 6, "Temperature vs. Power Program"(Pg 42 of 46)
3. Source: INPO Bank - Q# 23848 - Used at Susquehanna 1, 08/01/2002

NRC K/A System/E/A

System

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.47 **RO** 3.4 **SRO** 3.7 **CFR Link** (CFR: 41.10,43.5 / 45.12)

Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.

83 ☐ RO ☑ SRO Question ID: 0071509 Origin: Bank ☑ Memory Level

Given the following conditions exist after a reactor trip while implementing EOP 2526, "Reactor Trip Recovery":

- Pressurizer level 25% and lowering slowly
- RCS pressure 2230 psia and trending up slowly
- RCS Tavg 534°F and steady
- "A" SG level 41% and trending up slowly
- "B" SG level 55% and lowering slowly

Identify the procedure which will be implemented next and the step that may be performed out of its given sequence within that procedure?

A EOP-2526, "Reactor Trip Recovery", manually adjust steam generator feed flows to control SG levels ☐

B EOP-2526, "Reactor Trip Recovery", manually adjust charging and letdown to control pressurizer level ☑

C EOP-2536, "Excess Steam Demand Event", manually operate heaters and spray to control pressurizer pressure ☐

D EOP-2536, "Excess Steam Demand Event", manually operate steam dump/bypass valves to control RCS Tcold ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: SG levels are within band and do not require action to correct or preserve the safety function.
VALID DISTRACTOR: EOP-2526 is the correct procedure.

CHOICE (B) - YES
Step 1.10.2 of the EOP User's Guide provides the following two conditions when EOP steps may be performed out of the order listed in the procedure: 1) steps which are asterisked may be brought forward to correct or preserve a safety function, and 2) steps may be performed out of order after they have been accomplished once, if they are Continuously Applicable step as indicated by an asterisk. Pressurizer level is outside of the band given and trending in away from the band.

CHOICE (C) - NO
WRONG: EOP-2536 is not the correct procedure.
VALID DISTRACTOR: Manual control of pressurizer pressure is an asterisked step.

CHOICE (D) - NO
WRONG: RCS Tcold is within the band and does not require action to correct or preserve the safety function.
VALID DISTRACTOR: Manual control of Tcold via steam dup/bypass valves is an asterisked step.

References

1. OP-2260, "Unit 2 EOP Users Guide", Revision 8 (7/11/02), Section 1.10.2 (Pg 11 of 32)

NRC K/A System/E/A

System E02 Reactor Trip Recovery

Number EA2.2 **RO** 3.0 **SRO** 4.0 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret adherence to appropriate procedures and operation within the limitations in the facility's license and amendments as they apply to the Reactor Trip Recovery.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

84 ☐ RO ☑ SRO Question ID: 5000043 Origin: New ☐ Memory Level

The unit is operating at 100% power, equilibrium conditions. RCS leakage has been steady (confirmed by manual calculation) over the last 2 days at 0.98 gpm, of which 0.01 gpm is known tube leakage on #1 SG.

The RO makes a 60 gallon water addition as measured by Flow Integrator F-210X to the RCS during the 4 hour period of today's PPC leakrate calculation.

If Flow Integrator F-210X has inaccurately measured 5 gallons more than was actually injected, and the resulting leak rate change is attributed entirely to SG #1 tube leakage, then the new SG #1 tube leak rate based on PPC calculation will be ____________________

A 0.0308 gpm, which is less than the TS LCO for primary-to-secondary leakage. ☑

B 0.218 gpm, which is less than the TS LCO for primary-to-secondary leakage. ☐

C 0.0308 gpm, which exceeds the TS LCO for primary-to-secondary leakage and exceeds the pressure boundary leakage limit. ☐

D 0.218 gpm, which exceeds the TS LCO for primary-to-secondary leakage and meets the pressure boundary leakage limit. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(2) Facility operating limitations in the technical specifications and their bases.

CHOICE (A) - YES
5 gallons less added to RCS than actually added. Will be calculated as an increase of leakage of 5 gallons over 4 hours. This equates to a calculated increase in leakage of 0.0208 gpm. Added to existing SG tube leakage, new calculated pri-to-sec is 0.0308 gpm. TS limit is 0.035 gpm.

CHOICE(B) - NO
WRONG: Measured leakrate is 0.0308 gpm.
VALID DISTRACTOR: Plausible value, off by factor of 10.

CHOICE (C) - NO
WRONG: Measured leakrate does not exceed the TS LCO.
VALID DISTRACTOR: Applicant may think this exceeds SG and pressure boundary limits

CHOICE (D) - NO
WRONG: Measured leakrate does not exceed the TS LCO.
VALID DISTRACTOR: Could determine leakage to be 0.218 gpm, making this choice credible.

References

1. TS 3.4.6.2, "Reactor Coolant System Leakage", Amendment 228 (Pg 3/4 4-9)
2. SP-2602A, "Reactor Coolant Leakage", Revision 5 (08/31/04) (Pg 3 of 20)

NRC K/A System/E/A

System 022

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.12 **RO** 3.0 **SRO** 3.4 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of surveillance procedures.

85 ☐ RO ☑ SRO Question ID: 5000044 Origin: New ☑ Memory Level

In EOP-2528, "Loss of Offsite Power/Loss of Forced Circulation", a note states that a cooldown rate of 30°F/hr should be observed when RCS Tcold is below 230°F. This note is based on ___________________________.

A preventing uncoupling of the core and loops ☐

B ensuring a margin of safety against brittle failure ☑

C preventing a cold water accident following RCP restart ☐

D ensuring no void formation due to upper head temperature ☐

Justification

SRO ONLY QUESTION - Samples 55.43(2) Facility operating limitations in the technical specifications and their bases.

CHOICE (A) - NO
WRONG: During natural circulation, the maximum cooldown rate at which the RCS loops can be maintained coupled is dependent on decay heat and RCS temperature . Due to this, the cooldown rate must be lowered as the cooldown progresses. A cooldown rate of 30 to 60 degrees an hour should be maintainable initially, and a rate of 10 to 25 degrees per hour should be sustainable until RCS temperature reaches 300°F. While coupling is a concern, the higher limit of 30°F/hr is a technical specification requirement related to brittle fracture concerns.
VALID DISTRACTOR: Uncoupling is a concern during a natural circulation cooldown.

CHOICE(B) - YES
Cooldown rate is maintained in accordance with Technical Specifications to ensure a margin of safety against brittle (non-ductile) failure.

CHOICE (C) - NO
WRONG: Rate of cooldown does not affect RCP restart.
VALID DISTRACTOR: Plausible that stagnant legs can develop during natural circulation cooldown.

CHOICE (D) - NO
WRONG: Cooldown limit is a technical specification brittle fracture concern.
VALID DISTRACTOR: Void formation is a concern during natural circulation.

References

1. EOP-2528, "Loss of Offsite Power/Loss of Forced Circulation", Revision 15 (2/27/01), (Page 21 of 36)
2. TS Basis 3/4.4.9, "RCS Pressure and Temperature Limits", Amendment 218 (Pg B 3/4 4-5, 4-6)

NRC K/A System/E/A

System 026

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.20 **RO** 3.3 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 45.13)

"Knowledge of operational implications of EOP warnings, cautions, and notes."

86   ☐ RO ☑ SRO   Question ID: 5000045   Origin: Mod   ☑ Memory Level

The plant has been operating at 100% power. An automatic reactor trip has been initiated by RPS. All CEA position indications show all rods are out of the core. The reactor trip breakers will not open at C-04 or locally and the CEDM feeder breakers will not open.

Operators implement the appropriate EOP. Identify the correct procedure entered and the procedurally identified criteria for verifying that the reactivity control acceptance criteria is met?

A EOP-2525, "Standard Post Trip Actions", emergency boration in progress ☑

B EOP-2525, "Standard Post Trip Actions", power dropping and negative SUR ☐

C EOP-2540A, "Functional Recovery of Reactivity Control", emergency boration in progress ☐

D EOP-2540A, "Functional Recovery of Reactivity Control", power dropping and negative SUR ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - YES
EOP-2525 provides contingency actions to trip the reactor from the control room by opening the MG set output breakers. EOP-2540A is not entered until all SFSC evaluated. Reactivity SF will be met before transition from EOP-2525. Criteria are to take actions, already taken in stem, to try to trip reactor. Additionally, if any rods stick out, the condition will be addressed within EOP-2525 by emergency boration.

CHOICE(B) - NO
WRONG: Reactivity SF in EOP-2525 does not require power dropping and negative SUR. If these conditions are not met, the contingency steps require actions to attempt to trip reactor. Emergency boration is a required criterion for the stuck CEAs.
VALID DISTRACTOR: Power dropping and negative SUR are required to be checked to determine if contingency steps should be performed.

CHOICE (C) - NO
WRONG: EOP-2525 will address the problem. EOP-2540A will not need to be implemented.
VALID DISTRACTOR: EOP-2540A addresses the loss of reactivity control safety function.

CHOICE (D) - NO
WRONG: Power decreasing and negative SUR are not the criteria for confirming reactor shutdown.
VALID DISTRACTOR: If EOP-2540A were the procedure to be implemented, power dropping and negative SUR would be criteria for the SF.

References

1. EOP-2525, "Standard Post Trip Actions", Revision 20 (2/27/01) (Pg 3 of 26)
2. OP-2260, "Unit 2 EOP Users Guide", Revision 8 (7/11/02)
3. Source: INPO Bank - Q# 24679 - Used at Seabrook 1, 5/30/2003

NRC K/A System/E/A

System 029 Anticipated Transient Without Scram (ATWS)

Number EA2.09 **RO** 4.4 **SRO** 4.5 **CFR Link** (CFR 43.5 / 45.13)

Ability to determine or interpret the following as they apply to a ATWS: Occurrence of a main turbine/reactor trip

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

87 ☐ RO ☑ SRO Question ID: 0053649 Origin: Bank ☑ Memory Level

A reactor trip has occurred.

During the performance of EOP 2525, "Standard Post Trip Actions", the following conditions are reported:

PPO:
- SJAE RM is alarming
- Pressurizer level is 15% and lowering
- Pressurizer pressure is 1850 psia and lowering
- NO other apparent problems

SPO:
- S/G Blowdown has isolated
- #1 S/G level is 9% and rising
- #2 S/G level is 34% and rising
- #1 FRV Bypass ~ 60% open
- #2 FRV Bypass ~ 30% open
- Both S/G levels are rising at the same rate
- No other apparent problems

Per EOP-2525, what actions must be taken with regard to steam generators?

A Secure feed to #1 S/G. ☐

B Secure feed to #2 S/G. ☐

C Feed SGs to maintain levels of 10 to 80%. ☐

D Feed SGs to maintain levels of 40 to 70%. ☑

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: #1 SG is the unaffected SG. Level will be controlled between 40 and 70%.
VALID DISTRACTOR: Plausible that #1 unnecessary for heat removal.

CHOICE(B) - NO
WRONG: Step 7 contingency has the affected SG controlled between 40 and 70%.
VALID DISTRACTOR: Plausible that #2 would be isolated because of the SG tube leak.

CHOICE (C) - NO
WRONG: Both SGs will be maintained between 40 and 70%.
VALID DISTRACTOR: Step 6 has the operator ensure at least one SG is between 10 and 80% as a heat sink for the reactor coolant system.

CHOICE (D) - YES
EOP 2525 step 7.b. contingency actions for SJAE radiation monitor unexplained activity requires that feed be throttled to maintain 40-70 % to the SG with the highest radiation readings. The highest radiation reading will be in the SG with the higher level/lowest feed flow. Step 6c restores level to 40 to 70%.

References

1. EOP-2525, "Standard Post Trip Actions", Revision 20 (2/27/01) (Pg 16,18 of 26)
2. "EOP-2525, Standard Post Trip Actions Technical Guide", Revision 20 (Pg 19 of 38)

NRC K/A System/E/A

System 038

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.23 **RO** 3.9 **SRO** 4.0 **CFR Link** (CFR: 45.2 / 45.6)

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

88 ☐ RO ☑ SRO Question ID: 0054362 Origin: Mod ☐ Memory Level

While operating at 100% power, a plant trip occurs. While carrying out EOP-2525, Standard Post Trip Actions, the operators observe the following plant conditions:

* All CEAs are inserted.
* All buses are energized.
* Pressurizer Level is 10%, lowering.
* Pressurizer Pressure is 1700 psia, lowering.
* Tavg is 505 °F, lowering.
* RCS subcooling is 100 °F, rising.
* Feeding both SGs with Main Feedwater.
* #1 SG level 15% and dropping.
* #2 SG level 42% and rising.
* #1 SG pressure 450psia and dropping.
* #2 SG pressure 650 psia and dropping.
* Containment pressure 1.5 psig, rising.
* SJAE rad monitor activity rising.
* #2 MSL rad monitor activity rising.
* NO rad monitors in alarm.

Which procedure will the operators implement next?

A EOP 2532, Loss of Coolant Accident ☐

B EOP 2534, S/G Tube Rupture ☐

C EOP 2536, Excess Steam Demand ☐

D EOP 2540, Functional Recovery ☑

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
VALID DISTRACTOR: Pressurizer pressure is lowering, pressurizer level is lowering.

CHOICE(B) - NO
WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
VALID DISTRACTOR: Pressurizer pressure is dropping, no containment rad monitor alarms.

CHOICE (C) - NO
WRONG: Multiple events (SGTR and ESDE) require entry into the functional recovery procedure.
VALID DISTRACTOR: #1 S/G level is dropping.

CHOICE (D) - YES
Multiple events are in progress (SGTR and ESDE with failure of MSI), requiring entry into the functional recovery procedure.

References

1. OP-2260, "Unit 2 SOP User's Guide", Revision 1 (7/11/02) (Pg 9,10 of 32)
2. EOP-2541, Appendix 1, "Diagnostic Flowchart", Revision 000 (10/2/03) (Pg 1 of 1)

NRC K/A System/E/A

System A11 RCS Overcooling

Number AA2.1 **RO** 2.9 **SRO** 3.3 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the (RCS Overcooling) Facility conditions and selection of appropriate procedures during abnormal and emergency operations.

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

89 ☐ RO ☑ SRO Question ID: 5000046 Origin: New ☐ Memory Level

The plant is operating at 100% power when a large steam line rupture occurs on the upstream side of #1 Main Turbine Stop Valve (SV-1). The reactor and turbine are manually tripped. While responding to the event, the operator attempts to close the MSIVs using handswitches on C05.

Both MSIVs remain open. Which of the following choices is required to address this problem?

A Direct local manual closure of both MSIVs per EOP-2525, Standard Post-Trip Actions. ☐

B Direct local manual closure of both MSIVs per EOP-2536, Excess Steam Demand Event. ☐

C Place MSIV Bottle-Up Panel Isolation Switches in ISOL per EOP-2525, Standard Post-Trip Actions. ☑

D Place MSIV Bottle-Up Panel Isolation Switches in ISOL per EOP-2536, Excess Steam Demand Event. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: EOP-2525 contingency for MSIVs fail to close does not direct local manual closure.
VALID DISTRACTOR: Plausible that local action may be required.

CHOICE (B) - NO
WRONG: EOP-2536 contingency for MSIVs fail to close does not direct local manual closure.
VALID DISTRACTOR: Plausible that local action may be required.

CHOICE (C) - YES
EOP-2525 Step 6 contingency directs use of bottle-up panel switches.

CHOICE (D) - NO
WRONG: Operator would take action to operate bottle-up panel switches in EOP-2525 before entering EOP-2536.
VALID DISTRACTOR: Plausible that EOP-2536 would cover the necessary contingencies to address an MSIV fail to close.

References

1. EOP-2525, "Standard Post Trip Actions", Revision 20 (2/22/01) (Pg 13 of 26)

NRC K/A System/E/A

System E05

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.49 **RO** 4.0 **SRO** 4.0 **CFR Link** (CFR: 41.10 / 43.2 / 45.6)

Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

90 ☐ RO ☑ SRO Question ID: 5000047 Origin: New ☑ Memory Level

Refueling is in progress on Unit 2. During normal rounds, the Aux Building PEO reports that the red light on the SFP SW Area Radiation Monitor (RM-8139) local module is illuminated.

Which of the following is a possible reason for the reported indication?

A loss of power to the radiation monitor ☐

B local horn silence switch in the OFF position ☐

C actual high radiation condition in spent fuel pool area ☑

D Fuel Area Radn AEAS switch at ESF sensor cab in INHIBIT ☐

Justification

SRO ONLY QUESTION - Samples 55.43(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

CHOICE (A) - NO
WRONG: Loss of power extinguishes the light.
VALID DISTRACTOR: Plausible that function of red light is to indicate a loss of power.

CHOICE (B) - NO
WRONG: Local horn silence keyswitch disables the audible alarm but leaves the red light lit.
VALID DISTRACTOR: Plausible that function of red light is to inform that audible is defeated.

CHOICE (C) - YES
Local red light illuminates on sensed high radiation condition at a reading exceeding 50mR/hr.

CHOICE (D) - NO
WRONG: Keyswitch at ESF sensor cabinet functions to inhibit the trip and change logic from 2 /4 to 2 out of 3.
VALID DISTRACTOR: Plausible that red light designed to provide local indication of defeated input to ESAS.

References

1. RMS-00-C, "Radiation Monitoring System" Lesson, Revision 6 (9/18/02) (Pg 14,16 of 109)

NRC K/A System/E/A

System 061 Area Radiation Monitoring (ARM) System Alarms

Number AA2.01 **RO** 3.5 **SRO** 3.7 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Area Radiation Monitoring (ARM) System Alarms:
ARM panel displays

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

91   ☐ RO   ☑ SRO   Question ID: 5000048   Origin: New   ☐ Memory Level

Plant is stable at 100% power with no activities in progress and no equipment out of service when the FIRE SYSTEM TROUBLE (AB-19, C-06/7) and the `A' DIESEL ROOM (Zone 12, C-26) alarms are received. Approximately 1 minute later the following conditions are confirmed:

U3 Electric Fire Pump is stopped in standby
U3 Diesel Fire Pump is stopped in standby
U2 Electric Fire Pump is stopped in standby
Jockey Fire Pump is running
Panel C-26H ('A' DG local fire panel) alarm light lit
Panel C-26H one heat detector light lit
Panel C-26H audible alarm horn actuated
Other Panel C-26H indications are normal
No other control room alarms have actuated

The AB PEO has not entered the DG room.

This situation indicates a __________________.

A failure of the heat detector, go to ARP-2590I, "C-26 Alarm Response" ☑

B fire in the diesel generator room, go to AOP-2559, "Fire" ☐

C melted fusible link in deluge system, go to RP-16, "Trouble Reporting" ☐

D loss of ventilation, go to OP-2315E, "Diesel Generator Ventilation System" ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - YES
Actual fire would melt fusible link, causing supervisory air low pressure alarm.

CHOICE (B) - NO
WRONG: Fire would open deluge valve, resulting in additional alarms.
VALID DISTRACTOR: Plausible that detector has reported a valid condition.

CHOICE (C) - NO
WRONG: Melted fusible link would initiate spray flow, actuating deluge valve opening alarm.
VALID DISTRACTOR: Plausible that alarm caused by melted link.

CHOICE (D) - NO
WRONG: Temperature Switch TS-8435 provides a Diesel Gen 12U Room Temp Hi/Lo alarm on C-08 at 110°F.
VALID DISTRACTOR: Plausible that room temperature increase has caused the alarm.

References

1. FPS-04-C, "Fire Protection System" Lesson, Revision 3 (Pg 28,29,52 of 82

NRC K/A System/E/A

System 067 Plant fire on site

Number AA2.09 **RO** 2.4 **SRO** 2.7 **CFR Link** (CFR: 43.5 / 45.13)

Ability to determine and interpret the following as they apply to the Plant Fire on Site: That a failed fire alarm detector exists

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

92 ☐ RO ☑ SRO Question ID: 5000049 Origin: New ☑ Memory Level

4 days ago the reactor was manually tripped from 100% power because of a feed control malfunction. The problem has been corrected and a reactor startup is in progress. Current conditions are as follows:

- Tcold 532°F
- Pressurizer Pressure 2250 psia
- All RCPs in operation
- Reactor is critical
- Power is stabilized at 1E-3% while recording critical data

'B' RCP upper motor guide bearing temperature is observed to be 195°F and rising at 1°F every 3 minutes. Which of the choices below is immediately required?

A 'B' RCP will remain running, implement OP-2202, "Reactor Startup IPTE". ☐

B 'B' RCP will have to be stopped, implement EOP-2525, "Standard Post Trip Operations". ☑

C 'B' RCP will remain running, implement ARP-2590B-124, "RCP B Upper Guide Temp Hi". ☐

D 'B' RCP will have to be stopped, implement TS 3.4.1, "Coolant Loops and Coolant Circulation" ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: ARP-2590B provides necessary guidance. RCP must be stopped.
VALID DISTRACTOR: May not know the temperature limit on RCP upper guide bearing.

CHOICE (B) - YES
ARP directs trip of reactor/turbine, stop of RCP and reference to EOP-2525 at >194°F guide bearing temperature.

CHOICE (C) - NO
WRONG: Bearing temperature is above 194°F. ARP requires immediate trip of reactor/turbine and RCP.
VALID DISTRACTOR: May think bearing temperature is below maximum allowed value. ARP does provide necessary guidance on high temperature.

CHOICE (D) - NO
WRONG: TS 3.4.1 for MODE 3 only requires one RCP in operation. No TS impact to stopping RCP.
VALID DISTRACTOR: May think TS LCO will require action with less than 4 RCPs in operation.

References

1. ARP-2590B-124, "RCP B Upper Guide Temp Hi", Revision 0 (3/4/04)

NRC K/A System/E/A

System 003 Reactor Coolant Pump System (RCPS)

Number A2.03 **RO** 2.7 **SRO** 3.1 **CFR Link** (CFR: 41.5 / 43.5/ 45.3 / 45/13)

Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

93 ☐ RO ☑ SRO Question ID: 5000050 Origin: New ☑ Memory Level

The unit is in a refueling outage. A modification was installed on the condenser air removal system to troubleshoot air in-leakage problems..

If this installation was performed as a "temporary modification", what time limitation is associated with the modification?

A Unless required more frequently by SORC, the modification shall be audited within 90 days after installation and at least once per calendar quarter thereafter. ☐

B Unless required more frequently by SORC, the modification shall be audited within 90 days after installation and at least once per calendar year thereafter. ☐

C Unless authorized by Station Director, the modification shall be removed prior to the end of the next refuel outage or a time not to exceed 18 months, whichever is shorter. ☐

D Unless authorized by Station Director, the modification shall be removed prior to the end of the next refuel outage or a time not to exceed 24 months, whichever is shorter. ☑

Justification

SRO ONLY QUESTION - Samples 55.43(3) Facility licensee procedures required to obtain authority for design and operating changes in the facility.

CHOICE (A) - NO
WRONG: Auditing is required monthly.
VALID DISTRACTOR: Package processing is required within 90 days.

CHOICE (B) - NO
WRONG: Auditing is required monthly and additional requirements are imposed every six months.
VALID DISTRACTOR: Completed index sheets are retained until all temporary modifications for a respective calendar year are restored.

CHOICE (C) - NO
WRONG: The modification removal requirement is to not exceed 24 months.
VALID DISTRACTOR: "R" for refueling is defined as every 18 months in Tech Specs.

CHOICE (D) - YES
Requirement is as written.

References

1. WC-10, "Temporary Modifications", Revision 5 (5/26/04) (Pg 45 of 52)

NRC K/A System/E/A

System 055

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.20 **RO** 2.2 **SRO** 3.3 **CFR Link** (CFR: 43.5 / 45.13)

Knowledge of the process for managing troubleshooting activities.

 # 94 ☐ RO ☑ SRO Question ID: 5000051 Origin: New ☐ Memory Level

The unit is operating normally at 100% power when the following indications are noted:

- pressurizer pressure 2180 psia and decreasing
- pressurizer level 68% and increasing slowly
- containment atmosphere process radiation levels increasing
- pressurizer relief tailpipe temperatures at ~135°F and steady
- NO acoustic monitor alarms

Given these indications and assuming NO operator action has been taken, identify the problem and the required action.

A reactor coolant liquid leak, enter EOP-2525, "Standard Post Trip Actions" ☐

B pressurizer steam space leak, enter AOP-2568, "Reactor Coolant System Leak" ☑

C leaking power operated relief valve, enter ARP-2590B-043, "PORV RC-402 Open" ☐

D secondary steam system leak in containment, enter AOP-2575, "Rapid Downpower" ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Liquid leak cannot account for the increasing pressurizer level.
VALID DISTRACTOR: Given an uncontrollable RCS leak, the operator would have to trip the reactor and respond per EOPs.

CHOICE (B) - YES
Indications are consistent with a pressurizer steam space leak.

CHOICE (C) - NO
WRONG: PORV leakage would result in tailpipe temperature consistent with an isenthalpic expansion across the PORV. Slightly elevated tailpipe temperature is caused by steam leak in vicinity of temperature sensor.
VALID DISTRACTOR: PORV leakage would result in elevated tailpipe temperature.

CHOICE (D) - NO
WRONG: Secondary steam leak should not result in elevated radiation levels
VALID DISTRACTOR: A rapid downpower could reasonably be attempted.

References

1. AOP-2568, "Reactor Coolant System Leak", Revision 7 (5/27/03)

NRC K/A System/E/A

System 007

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.1 Conduct of Operations

Number 2.1.7 **RO** 3.7 **SRO** 4.4 **CFR Link** (CFR: 43.5 / 45.12 / 45.13)

"Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation."

95

☐ RO ☑ SRO Question ID: 5000052 Origin: New ☑ Memory Level

An automatic reactor trip and SIAS have occurred while operating at power.

30 minutes into the event, the following conditions exist:
- Crew is performing EOP-2532 "Loss of Coolant Accident"
- CET temp 320°F
- pressurizer pressure 28 psia
- containment pressure 9 psig
- SIAS, CIAS, EBFAS, MSI and CSAS have actuated
- all equipment is functioning per design

After several RWST LEVEL CH LO/LO alarms are received, the operator determines that SRAS has NOT actuated. Choose the NEXT correct action in response to the SRAS failure.

A Manually stop both Charging pumps. ☐

B Manually close both Gravity Feed valves (CH-508/509). ☐

C Manually stop both Low Pressure Safety Injection pumps. ☑

D Manually close both RWST header outlet valves (CS-13.1A/B). ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Charging pumps stopped after ensuring automatic actions of SRAS
VALID DISTRACTOR: Charging pumps are stopped after SRAS

CHOICE (B) - NO
WRONG: Gravity feed valves are closed after ensuring automatic actions of SRAS
VALID DISTRACTOR: Gravity feed valves are closed after SRAS

CHOICE (C) - YES
Directed in Step 47 of EOP-2532.

CHOICE (D) - NO
WRONG: RWST outlet valves closed after ensuring automatic actions of SRAS
VALID DISTRACTOR: RWST outlet valves closed after SRAS

References

1. EOP-2532, "Loss of Coolant Accident", Revision 23 (3/31/04) (Pg 39, 40 of 95)

NRC K/A System/E/A

System 026 Containment Spray System (CSS)

Number A2.02 **RO** 4.2* **SRO** 4.4* **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.13)

Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control,or mitigate the consequences of those malfunctions or operations: Failure of automatic recirculation transfer

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

96 ☐ RO ☑ SRO Question ID: 0054565 Origin: Bank ☑ Memory Level

Following a plant trip, the Emergency Diesel Generators (EDG's) are supplying their respective Bus 24C and 24D due to a failure to transfer to the Reserve Station Service Transformer (RSST).

While still in EOP-2528, "Loss of Offsite Power/Loss of Forced Circulation", the RSST is now available to supply Facility 1 electrical loads. NO electrical faults exist.

Based on these conditions, which of the following statements identifies the procedure and the CORRECT sequence of steps needed to restore Bus 24C to a normal post-trip alignment?

A Per EOP-2541, Appendix 23, "Restoring Electrical Power", reset ESAS UV signal, parallel RSST to 24C, open D/G breaker, close Bus 24A-24C tie breaker. ☑

B Per AOP-2502C, "Loss of Vital 4.16 kV Bus 24C", reset ESAS UV signal, parallel RSST to 24C, open D/G breaker, close Bus 24A-24C tie breaker. ☐

C Per EOP-2541, Appendix 23, "Restoring Electrical Power", reset ESAS UV signal, parallel RSST to 24C, close Bus 24A-24C tie breaker, open D/G breaker. ☐

D Per AOP-2502C, "Loss of Vital 4.16 kV Bus 24C", reset ESAS UV signal, parallel RSST to 24C, close Bus 24A-24C tie breaker, open D/G breaker. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - YES
These actions are directed in the listed sequence in Appendix 23.

CHOICE (B) - NO
WRONG: AOP-2502C does not provide step sequence for re-energizing from normal source.
VALID DISTRACTOR: AOP-2502C provides steps for energizing bus from the emergency diesel generator.

CHOICE (C) - NO
WRONG: Sequence is not correct. Tie breaker is not closed until after D/G breaker is open.
VALID DISTRACTOR: Plausible to think tie breaker should be closed before opening D/G breaker.

CHOICE (D) - NO
WRONG: AOP-2502C does not provide step sequence for re-energizing from normal source.
VALID DISTRACTOR: Plausible to think that AOP would provide specific guidance for transferring bus back to normal source.

References

1. OP-2343, "4160 Volt Electrical System", Revision 20 (9/9/04), Section 4.20 "Restoring Bus 24C to Unit 2 RSST with Emergency Diesel Generator
Supplying" (Pg 42 of 71)

NRC K/A System/E/A

System 062

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.2 **RO** 4.0 **SRO** 3.5 **CFR Link** (CFR: 45.2)

Ability to manipulate the console controls as required to operate the facility between shutdown and designated power levels.

97 ☐ RO ☑ SRO Question ID: 5000053 Origin: New ☐ Memory Level

Which of the following describes a condition required to be reported to the NRC under 10CFR50.72 and the correct time limit for reporting?

A 1 hour report due to deviation from the plant Technical Specifications authorized pursuant to 10CFR50.54(x) ☑

B 1 hour report due to a condition that could have prevented fulfillment of a safety function needed to mitigate consequences of an accident ☐

C 4 hour report due to failure to perform required surveillance test within technical specification allowable time limits ☐

D 4 hour report due to the nuclear power plant being in an unanalyzed condition that significantly degrades plant safety ☐

Justification

SRO ONLY QUESTION - Samples 55.43(1) Conditions and limitations in the facility license.

CHOICE (A) - YES
10CFR50.72(b)(1) requires a 1 hour ENS notification if provisions of CFR50.54(x) invoked.

CHOICE (B) - NO
WRONG: Fulfillment of a safety function is an 8 hour notification under 10CFR50.72(b)(3)(v).
VALID DISTRACTOR: Plausible that safety function needed for accident mitigation would be a 1 hour report.

CHOICE (C) - NO
WRONG: Failure to perform a surveillance test within allowable time limits is not 1, 4 or 8 hour reportable.
VALID DISTRACTOR: Plausible that missed surveillance would be reportable as a violation of technical specification requirements.

CHOICE (D) - NO
WRONG: 10CFR50.72(b)(3) requires an 8 hour notification.
VALID DISTRACTOR: Plausible that an unanalyzed condition would require a 4 hour report.

References

1. RAC-14, "Non-Emergency Station Events", Revision 1 (3/24/04), Attachment 1 (Sheet 1, 2, 3 of 17)

NRC K/A System/E/A

System 076

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.4 Emergency Procedures /Plan

Number 2.4.30 **RO** 2.2 **SRO** 3.6 **CFR Link** (CFR: 43.5 / 45.11)

Knowledge of which events related to system operations/status should be reported to outside agencies.

98 ☐ RO ☑ SRO Question ID: 5000054 Origin: Bank ☐ Memory Level

On your shift, a monthly surveillance item is discovered overdue. Required due date was November 28th. Assume today is December 3rd and the performance of the Surveillance Test has begun. The previous surveillance tests for this component/system were Due and Completed as shown below.

Due Date:	Completed Date:
- August 25	- August 28
- September 28	- October 1
- November 1	- October 28

Which ONE of the following statements describes the status of the component/system and the justification for that status?

A The component/system is INOPERABLE because 3.25 times the time interval for three consecutive tests has been exceeded. ☐

B The component/system is INOPERABLE because more than 24 hours have elapsed from the due date plus the allowed extension. ☐

C The component/system is OPERABLE because the Technical Specifications allow time from previous tests to be carried forward. ☐

D The component/system is OPERABLE because the Technical Specifications allow a time extension which has not been exceeded. ☑

Justification

SRO ONLY QUESTION - Samples 55.43(2) Facility operating limitations in the technical specifications and their bases.

CHOICE (A) - NO
WRONG: 25% extension still allows time to perform the surveillance.
VALID DISTRACTOR: Plausible that consecutive test frequency is to be taken into account.

CHOICE (B) - NO
WRONG: Extension allows for 7 days from due date to perform test. Only 5 days have elapsed
VALID DISTRACTOR: TS 4.0.3 allows a 24 hour extension from time discovered missed surveillance.

CHOICE (C) - NO
WRONG: TS 4.0.2 basis states "it is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages."
VALID DISTRACTOR: Plausible to assume that testing frequency includes consideration of previous test history.

CHOICE (D) - YES
Per TS 4.0.2, each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance time interval. Per TS 4.0.3, if it is discovered that a Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater.

References

1. Technical Specifications, Section 3/4.0, "Applicability", Amendment 271 (Pg 3/4 0-2)
2. Source: INPO Bank - Q# 22908 - Used at DC Cook 1, 12/9/2002

NRC K/A System/E/A

System 002

Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.12 **RO** 3.0 **SRO** 3.4 **CFR Link** (CFR: 41.10 / 45.13)

Knowledge of surveillance procedures.

 # 99 ☐ RO ☑ SRO Question ID: 5000055 Origin: New ☐ Memory Level

The unit is at 100% power. SG LEVEL SETPOINT DEVIATION HI/LO alarm is received (Panel C-05, D-16). The operator reports an apparent failure low of #1 Steam Generator Pressure Transmitter, PT-4243 (steam flow pressure compensation channel).

What automatic plant response is expected and what action is required?

A SG level has decreased, per OP-2385, "FEEDWATER CONTROL SYSTEM OPERATION", take manual FRV control, select alternate steam pressure channel, return FRV to auto. ☐

B SG level has increased, per OP-2385, "FEEDWATER CONTROL SYSTEM OPERATION", take manual FRV control, select alternate steam pressure channel, return FRV to auto. ☐

C SG level has decreased, per ARP-2590D-064, "SG LEVEL SETPOINT DEVIATION HI/LO", take manual FRV control, maintain level in manual until pressure transmitter repaired. ☑

D SG level has increased, per ARP-2590D-064, "SG LEVEL SETPOINT DEVIATION HI/LO", take manual FRV control, maintain level in manual until pressure transmitter repaired. ☐

Justification

SRO ONLY QUESTION - Samples 55.43(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

CHOICE (A) - NO
WRONG: Alternate channel unavailable due to pressure compensation failure.
VALID DISTRACTOR: A loss of one channel input typically requires selection of an alternate channel.

CHOICE (B) - NO
WRONG: OP-2385 does not provide the guidance for response to a failed channel.
VALID DISTRACTOR: Plausible that OP-2385 would provide the necessary guidance.

CHOICE (C) - YES
PT4243 provides pressure compensation to both control channel steam flow instruments on #1 MSL. A failure low will reduce steam flow on both channels, resulting in the 3-element control system throttling closed FRVs. Level will decrease. ARP-2590D-064 directs FRV control in manual. Would have to maintain FRV in manual until pressure compensation restored.

CHOICE (D) - NO
WRONG: Level will decrease because the control system will percieve a reduction in steam flow.
VALID DISTRACTOR: May think that low compensation pressure will result in higher indicated steam flow.

References

1. ARP-2590D-064, "SG LEVEL SETPOINT DEVIATION HI/LO", Revision 000 (2/12/04)
2. 25203-26002, "MAIN STEAM FROM GENERATORS" SH.1 OF 5, Revision 57 (3/7/02) (K-8)

NRC K/A System/E/A

System 035 Steam Generator System (S/GS)

Number A2.03 **RO** 3.4 **SRO** 3.6 **CFR Link** (CFR: 41.5 / 43.5 / 45.3 / 45.5)

Ability to (a) predict the impacts of the following malfunctions or operations on the GS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Pressure/level transmitter failure

NRC K/A Generic

System

Number **RO** **SRO** **CFR Link**

 # 100 ☐ RO ☑ SRO **Question ID: 0053324** **Origin: Mod** ☐ Memory Level

You are the Fuel Handling SRO. The refueling cavity is filled to 36' 8" and core alterations are in progress on Unit 2. The upper manways on SG #2 are off for inspection of the can decks. Both LPSI Pumps are running on SDC.

Which of the following activities CANNOT be authorized?

A remove pressurizer power operated relief valve ☐

B disassemble #2 main steam isolation valve ☑

C shutdown 'A' LPSI and place in standby ☐

D lower refueling water storage tank level by 5% ☐

Justification

SRO ONLY QUESTION - Samples 55.43(7) Fuel handling facilities and procedures.

CHOICE (A) - NO
WRONG: PORV removal does not affect refueling or containment integrity.
VALID DISTRACTOR: Plausible that work could not be authorized during core alterations.

CHOICE (B) - YES
With manways removed, disassembly of #2 MSIV allows a direct path to atmosphere. Containment integrity is required when performing core alterations or moving irradiated fuel in Containment.

CHOICE (C) - NO
WRONG: Only 1 LPSI required to be in operation with full refuel pool.
VALID DISTRACTOR: Plausible that not allowed to stop LPSI pump during core alterations.

CHOICE (D) - NO
WRONG: Core alterations do not impose any limit on RWST level when refuel pool is filled.
VALID DISTRACTOR: Plausible that limits are imposed on RWST level during core alterations.

References

1. SP-2614B-002, "Containment Closure With SG Secondary Side Open to Containment", Revision 3 (4/20/01) (Pg 3 of 3)
2. Technical Specification 3.9.4, "Containment Penetrations", Amendment 245 (Pg 3/4 9-4)

NRC K/A System/E/A

System
Number **RO** **SRO** **CFR Link**

NRC K/A Generic

System 2.2 Equipment Control

Number 2.2.18 **RO** 2.3 **SRO** 3.6 **CFR Link** (CFR: 43.5 / 45.13)

Knowledge of the process for managing maintenance activities during shutdown operations.