1 ☑ RO ☑ SRO Question ID: 5000001 Origin: New

The plant was tripped from 100% power due to a report of a large steam leak in the East Penetration Area, upstream of MSIV MS-64A which has filled the area with steam. Both MSIVs were manually closed immediately following the trip at the direction of the CRS. 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

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

What is the appropriate action to take?

A Ensure MSI is actuated. ☐

B Manually initiate SIAS. ☐

C Stop RCPs as necessary. ☑

D Close PORV Block valves. ☐

Justification

CHOICE (A) - 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 (B) - NO
WRONG: EOP-2525, "Standard Post Trip Actions", directs the operator to ensure SIAS has actuated if pressurizer pressure is less than 1714 psia. Pressurizer pressure, at 1800 psia, is still well above 1714 psia.
VALID DISTRACTOR: an applicant could decide a manual SIAS is needed after incorrectly diagnosis the event as a pressurizer steam space leak..

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

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

- **A** Only the LBLOCA clears the 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

A loss of coolant accident occurs from 100% power. Following the reactor trip, an SPDS Safety Function display screen shows a quality tag of "U" to the left of the estimate value for Loop 1 cold leg temperature (Tc). The associated sensor data screen displays an "R" to the right of one of the cold leg temperature inputs.

Given these indications, the Tc estimate is based on input ____________

A nearing the calibrated range, treat as questionable. ☑

B exceeding error bounds, not acceptable for use. ☐

C by a validated estimate, may be treated as reliable. ☐

D from a redundant sensor, acceptable for use. ☐

Justification

CHOICE (A) - YES
Quality Tag "U" means the estimate is based on fewer than the optimum number of inputs, or one or more of the inputs used is tagged R. For example, a calculation using four inputs might be based on only three, or an average using two inputs might be based on one "good" value and one "R" value. An un-validated estimate should be treated as questionable. Operating personnel should attempt to validate the estimate using available control panel instruments.

CHOICE (B) - NO
WRONG: The input is flagged with a Quality Tag of "F" if it exceeds its error bounds. Values tagged S, X. or F are MAGENTA, and are not used in calculating Safety Function estimates. The Safety Function estimate Quality Tag would be labeled "N" if the number of usable inputs would not support development of an estimate. It would be unlabeled or blank if a sufficient number of inputs remained to allow calculation of the estimate.
VALID DISTRACTOR: an applicant may misinterpret the Quality Tag as indicative of an input exceeding error bounds.

CHOICE (C) - NO
WRONG: A validated estimate is one where a sufficient number of "good" inputs are available for calculation of the estimate. A validated estimate may be treated as reliable.
VALID DISTRACTOR: an applicant may interpret the Quality Tags as indicating the estimate has been validated.

CHOICE (D) - NO
WRONG: "R" stands for "range checked", not "redundant".
VALID DISTRACTOR: an applicant may be unaware of the meaning of the "R" Quality Tag.

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

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
lower - 790
middle - 710
upper - 680

"B" Pump
lower - 180
middle - 470
upper - 1540

"C" Pump
lower - 810
middle - 180
upper - 1190

"D" Pump
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

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.

If auto makeup fails to start, the rate of VCT level decrease will be approximately ______ times the prior rate of pressurizer level decrease.

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

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

The following conditions exist on Unit 2:

o The reactor is shutdown
o Both trains of SDC in service
o RCS temperature is 300°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)

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

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

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 5° F or more lower than #2 loop Tc. ☐

B #1 loop Th 10° F or more higher than #2 loop Th. ☑

C #1 loop delta-P 5 psi or more less than #2 loop delta-P. ☐

D #1 SG pressure 50 psi or more greater 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

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 o mitigate this event?

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

B Place both SG Auto Permissive OVERRIDE/MAN/START RESET Switches to PULL-TO-LOCK. ☑

C Shift both Auxiliary Feedwater Regulating Valve Manual Loading Stations to MANUAL and CLOSED. ☐

D Momentarily shift #2 S/G Auxiliary Feedwater Regulating Valve RESET/NORM/OVRD Switch 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) - 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 (C) - 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 (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

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
- RCS subcooling => 98 deg F
- Thot => 584 deg F
- Tcold => 581 deg F
- #1 SG WR level => 35 inches
- #2 SG WR level => 22 inches
- 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

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 => 525 degrees F and constant
RCS Thot => 512 degrees F and going down
CET => 523 degrees F and going down
RCS Pressure => 1500 psia

☐

B RCS Tcold => 475 degrees F and constant
RCS Thot => 533 degrees F and going down
CET => 522 degrees F and going down
RCS Pressure => 1600 psia

☐

C RCS Tcold => 538 degrees F and going down
RCS Thot => 560 degrees F and going down
CET => 561 degrees F and going down
RCS Pressure => 1380 psia

☐

D RCS Tcold => 538 degrees F and constant
RCS Thot => 550 degrees F and constant
CET => 549 degrees F and constant
RCS Pressure => 1720 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)

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

Operators are reducing reactor power at 5% per hour to remove 'B' SGFP from service for planned corrective maintenance on its turbine speed control system. Power level is at 692% when a transient is indicated by multiple alarms. The PPO observes that the 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 PO 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

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 ☐

B trip on overspeed on engine startup ☐

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

Plant is operating at 100% power. '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. Tech Spec Action Statement 3.7.2.1 has been entered for an inoperable 'B' Service Water loop. Long Island Sound water temperature is 37 degrees 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. Assuming no operator actions related to Service Water and considering conditions prior to the fault as "normal", identify the cause for the following indications:

PPC Points:
F6433 'A' RBCCW HX SW Flow => 5180 gpm
F6434 'B' RBCCW HX SW Flow => 1220 gpm
F6435 'C' RBCCW HX SW Flow => 5250 gpm

Local EDG SW Flows:
FIC-6397 'A' EDG => 1165 gpm
FIC-6389 'B' EDG => 145 gpm

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

B 'C' RBCCW HX SW flow is HIGHER than normal because of a leak at 'C' RBCCW HX SW inlet. ☑

C 'A' EDG SW flow is HIGHER than normal because of a leak at 'A' EDG Duplex Strainer SW inlet. ☐

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

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) - 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.

CHOICE (C) - 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 as expected during EDG operation. No leak is indicated.
VALID DISTRACTOR: Applicant may think the given flow is higher than normal due to a leak.

CHOICE (D) - 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.

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

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 the red '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 0.5 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: 0054381 Origin: Bank

The unit has experienced a reactor trip with the following conditions:

- VR-11 was lost at the time of the trip.
- All indicating lights for Group B CEA #6 on the Core Mimic display are out.
- Further investigation reveals no bulbs on the Core Mimic display are defective.
- The PPC display of Group B CEA #6 position reads 180.

Which one of the following conclusions can be made about the position of CEA #6 based on the given indications?

A It is stuck fully withdrawn. ☐

B It is only one (1) step withdrawn. ☐

C It is inserted at least 10 steps, but is NOT fully inserted. ☐

D It is not fully withdrawn, but is at least two (2) steps withdrawn. ☑

Justification

CHOICE (A) - NO
WRONG: CEA fully withdrawn would have a "Red" light lit because the magnet on the top of the extension shaft would close the UEL reed switch. (CED-01-C, Pg 30 of 67)
VALID DISTRACTOR: Applicant may think CEA is fully withdrawn because PPC indicates 180 steps.

CHOICE (B) - NO
WRONG: This indication is indicative of the Upper Electrical Limit (UEL) reed being opened (ie. the CEA is inserted >/= 1 step), but the Lower Electrical Limit is not lit (green light). Therefore, the CEA must be at least 2 steps withdrawn. (CED-01-C, Pg 30 of 67)
VALID DISTRACTOR: Applicant may think rod is only 1 step withdrawn.

CHOICE (C) - NO
WRONG: CEA inserted at least 10 steps could only be stated if the blue mimic light was lit by the PPC knowing the CEA is below the Exercise Limit. However, unless the LEL or dropped CEA light is lit (and stem states all lights are out) the PPC cannot tell the CEA location.
VALID DISTRACTOR: Applicant may think that CEA is inserted 10 steps.

CHOICE (D) - YES
This indication is indicative of the Upper Electrical Limit (UEL) reed being opened (ie. the CEA is inserted >/= 1 step), but the Lower Electrical Limit is not lit (green light). Therefore, the CEA must be at least 2 steps withdrawn.

References

1. CED-01-C, "Control Element Drive System" Lesson, Revision 4 (1/26/04) (Pg 28, 30, 55, 56, 57 of 67)

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

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 another 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

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?

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

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

The breaker for Reactor Coolant Pump 'B' trips open.

Assuming no operator action, compare SG1 with SG2 approximately 3 minutes after the breaker trip. For SG2 which of the following sets of conditions describes the expected conditions for the parameters listed below?

Steam Pressure	Steam Flow	Feed Flow
__________	__________	__________

A HIGHER, HIGHER, HIGHER ☐

B HIGHER, HIGHER, EQUAL ☑

C EQUAL, LOWER, LOWER ☐

D EQUAL, EQUAL, LOWER ☐

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 automatic reactor trip on low reactor coolant flow in Loop 1B (SG1) at a 92% flow setpoint. Two minutes after the event the post-trip decay heat will be removed via turbine bypass valves. Both loops will be steaming. However, Loop 1 has a lower primary flowrate and will therefore be transferring less energy to SG1 than Loop 2 is transferring to SG2 (Q-dot = m-dot * c-sub-p * delta-T). 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 SLIGHTLY HIGHER in SG2 because of the additional energy transfer. Steam flow could not be higher from SG2 if the pressures were equal.

CHOICE (C) - 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.

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)

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

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 degrees F

Given these conditions, RCP 1A must be tripped because of high _______________.

A motor vibration ☐

B seal bleed-off flow ☐

C oil reservoir level ☐

D winding temperature ☑

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: Seal bleed-off flow is approximately 1 gpm by design. The high flow alarm actuates at 2 gpm.
VALID DISTRACTOR: because the applicant may think that bleed-off flow is excessive.

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, perators 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

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

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?

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.

Source: Indian Point 3 NRC Exam, 12/2003

References

1. 10CFR20 "Standards for Protection Against Radiation", Subpart 20.1206, "Planned Special Exposures"
2. RPM 5.1.5, "Planned Special Exposures"

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

A steam generator tube rupture has just 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 blocking Main Steam Isolation signal ☐

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.

Source: Indian Point 3 NRC Exam, 12/2003

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)

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

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

D small oily rag bin fire in turbine building main condenser pit, 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.

Source: Indian Point 3 NRC Exam, 12/2003

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

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.

76 ☐ RO ☑ SRO Question ID: 0055020 Origin: Mod

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 expected actions of the CEDS?

A ONLY Group 6 CEAs will insert. ☐

B ONLY Group 7 CEAs will insert. ☐

C BOTH Group 6 and Group 7 CEAs will insert. ☐

D NEITHER Group 6 nor Group 7 CEAs will insert. ☑

Justification

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.

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.

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.

References

1. CED-01-C, "Control Element Drive System" Lesson, Revision 4 (1/26/04) (Pg 28, 30, 55, 56, 57 of 67)

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

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

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

The plant is at 100% when 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 => 3.7 psig
PT-8114 => 3.6 psig
PT-8115 => 3.7 psig
PT-8116 => 4.1 psig
PT-8117 => 4.0 psig
PT-8238 => 3.9 psig
PT-8239 => 3.5 psig

Which of the following statements and associated procedural action is correct about the "SIAS OR UV ACTUATION SIG CH 2 TRIP" annunciator on C-01?

A The alarm is valid, 2 of the associated Containment Pressure Channels are tripped, enter EOP-2525, "Standard Post Trip Actions". ☐

B The alarm is valid, 1 of the associated Containment Pressure Channels is tripped, enter EOP-2525, "Standard Post Trip Actions". ☐

C The alarm is not valid, 1 of the associated Containment Pressure Channels is tripped, enter the issue into the corrective action system. ☑

D The alarm is not valid, none of the associated Containment Pressure Channels is tripped, enter the issue into the corrective action 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) - 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.
VALID DISTRACTOR: Multiple pressure transmitters (8116, 8117, 8238) are reading above 3.8 psig.

CHOICE (B) - NO
WRONG: 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. 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 (C) - 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 chanel above the SIAS actuation setpoint. With less than 2 safety channels above setpoint, the alarm is not valid.

CHOICE (D) - NO
WRONG: PT-8116 is reading greater that 3,8 psig.
VALID DISTRACTOR: Alarm actuated by 2 of 4 ESF channels (8113 thru 8116) equal to or greater than 3.8 psig. PT-8116 is an ESF channel above the SIAS actuation setpoint.

Source: Indian Point 3 NRC Exam, 12/2003

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"

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

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 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 0600 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.

Source: INPO Bank - Q# 24702 - Used at Seabrook, 05/30/2003

References

1. Technical Specification 3/4.8.1, Pages 3/4.8-1a (Amendment 261), 3/4.8-2a (Amendment 277)

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.