Question #1

The plant is operating at 100% power.

- The Switchyard North Bus is lost.
- The Reactor subsequently trips.

What is the Alternating Current (AC) electrical supply path?

A. RSST to 25A, 25B, 24C and 24D
24C to 24A; 24D to 24B

B. “A” EDG to 24C and “B” EDG to 24D
24C to 24A; 24D to 24B

C. RSST to 25A, 25B, 24C and 24D
24C to 22E; 24D to 22F

D. “A” EDG to 24C and “B” EDG to 24D
24C to 22E; 24D to 22F

Question #1 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 007 EK2.09 Importance Rating: 3.5

K/A Statement: Knowledge of the relationship between a Reactor Trip and the following systems or components: AC distribution system

Proposed Answer: **D**

Justification: With the North Bus lost the unit 2 RSST is not available and on a plant trip the AC electrical busses will not transfer from the NSST to the RSST. This will result in the EDG automatically starting and energizing vital AC buses (24C, 24D, 22E, and 22F) only. Non-vital buses (25A, 25B, 24A, and 24B) will not be energized from the EDGs.

Plausibility:

A. The RSST to 25A, 25B, 24C and 24D with 24C to 24A and 24D to 24B is the normal supply when the RSST is available. This is plausible because this is what is expected on a normal plant trip. The candidate must know that the unit 2 RSST is powered from the north bus to not select this answer.
B. With the RSST not available the "A" EDG powers 24C and "B" EDG powers 24D. However, 24C does not power 24A and 24D does not power 24B. This is plausible because on a normal plant trip 24C powers 24A and 24D powers 24B.
C. The RSST to 25A, 25B, 24C and 24D with 24C to 22E and 24D to 22F is the normal supply when the RSST is available. This is plausible because this is what is expected on a normal plant trip. The candidate must know that the unit 2 RSST is powered from the north bus to not select this answer.
D. Correct.

Technical Reference(s): 1. In-House Electrical System Lesson Text (IHE-00-C, rev. 10/2)
2. 345 KV Lesson Text (rev. 8/1)
3. IHE Electrical Distribution System Figure
4. 345 KV Simple One-Line Figure

Provided reference(s): None

Learning Objective: 281520 ILT Given a loss of normal (LNP) has occurred, identify the sequenced response to restore power to a class 4160 VAC ESF bus.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-26, replace "and" in second part of anwsers with ";".

Changes completed 7-26.

Question #2

The plant is operating at 100% power when a Pressurizer Safety valve lifts and remains partially open.

- The Reactor trips on low Pressurizer pressure.
- Pressurizer level is 80% and rising.

1. Is Pressurizer level an accurate indication of RCS inventory?

   AND

2. Why?

A. 1. No
   2. RCS voiding results in increasing Pressurizer level.

B. 1. Yes
   2. RCP flow will sweep any voids from the RCS to the Pressurizer steam space.

C. 1. No
   2. Steam bubbles form in the reference legs of the level instruments.

D. 1. Yes
   2. Neither the reference leg nor the variable leg are affected by the open Safety.

Question #2 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 008 AK1.03 Importance Rating: 4.0

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Pressurizer Vapor Space Accident: Consequences due to a PZR vapor space leak.

Proposed Answer: A

Justification: The conditions in the stem will result in a bubble forming in the Reactor Vessel Head area. Indicated Pressurizer level is not an accurate indication of RCS inventory because of RCS voiding. This meets the K/A since how will Pressurizer level respond is knowledge of the operational implications and/or cause and effect relationships of a Pressurizer vapor space leak.

Plausibility:

A. Correct.
B. Incorrect. Pressurizer level is not accurate during a vapor space accident. It is plausible because a common misconception is that voids forming in the RCS will be transported by the Reactor Coolant pumps to the Pressurizer. However, once subcooling is lost, a bubble will form in the Reactor Vessel Head because this is the hottest temperature in the RCS (Head maintains a lot of heat).
C. Incorrect. Pressurizer level is not accurate during a vapor space accident is correct, but the reason is not due to steam bubble formation in the reference legs. It is plausible because Pressurizer level may be somewhat inaccurate as compared to normal plant conditions at 2250 psia.
D. Incorrect. Pressurizer level is not accurate during a vapor space accident. It is plausible due to the common misconception that the lowest pressure in the system is at the leak and therefore any flashing to steam would occur in the Pressurizer and not in the Reactor Vessel area.

Technical Reference(s): (1) EOP 2532, Loss of Coolant Accident Lesson (VISION 55700, rev. 5/0, Unit 2 ILT 2019-2020(, pages 1, 33, and 39.
(2) EOP 2532, Loss of Coolant Accident Tech Guide (rev. 29), pages 1 and 7.

Provided reference(s): None.

Learning Objective: 283785 ILT Describe the expected plant response to a Loss of Coolant Accident.

Source: Bank #451895

History: Millstone 2, 2014 NRC exam, Question 2.

Cognitive Level: Comprehension or Analysis

10CFR55.41(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: This was one of the pre-submittal questions the NRC provided feedback on June 3. Their comments were incorporated on Jun 6. Comments included; 1. Add to stem: “Pressurizer level is 80% and rising”, 2. revise stem question: “1. Is Pressurizer level accurate? AND 2. Why?, 3. Answer D: Correct the two 1’s. 4. remove from the justification “indicate subcooling has been lost”. In additions revised justification to align with question changes and shortened answers to align with stem. Jwr 6/6/2022. NRC feedback 7-26, was to make C.2 a complete sentence and make answer D.2 a reason that works better with “Yes”. Changes completed 7-29.

Question #3

The plant experienced a Small Break Loss of Coolant Accident.

- EOP 2532, Loss of Coolant Accident, is in progress.
- SIAS, CIAS, and EBFAS have actuated.
- All Circulating Water pumps are secured.

How does EOP 2532 direct the cool down be performed and monitored?

1. The ___1.___ will be used to depressurize the Steam Generators.

   AND

2. ___2.___ temperatures are used to monitor/control the cooldown rate.

A. 1. Condenser Steam Dump Valves
   2. CET

B. 1. Condenser Steam Dump Valves
   2. $T_{COLD}$

C. 1. Atmospheric Dump Valves
   2. CET

D. 1. Atmospheric Dump Valves
   2. $T_{COLD}$

Question #3 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 009 EA1.14 Importance Rating: 3.4

K/A Statement: Ability to operate and/or monitor the following as they apply to a Small-Break LOCA: Secondary pressure control.

Proposed Answer: **D**

Justification: The Atmospheric Dump Valves (ADVS) will be used to depressurize the Steam Generators and $T_{COLD}$ temperatures ($T_C$) are required to monitor the cool down. The ADVs must be used because the Steam Dump Valves (SDV) will not operate without vacuum in the condenser. And with no Circulating Water pumps operating Condenser vacuum is required to be removed (step 16.1). Monitoring with $T_{COLD}$ temperatures ($T_C$) is specified in the note prior to the cool down step (step 19). This meets the K/A since if covers ability to operate (ADVs to depressurize the Steam Generators) and monitor ($T_C$) for a Small Break LOCA.

Plausibility:

A. Incorrect. SDVs and CETs are not correct. It is plausible because the SDVs would be used if the condenser is available, and is the preferred method to cool down and CETs are used when not in forced flow or natural circulation.
B. Incorrect. SDVs is not correct. It is plausible because CETs are used when not in forced flow or natural circulation.
C. Incorrect. CETs is not correct. It is plausible because CETs are used when not in forced flow or natural circulation.
D. Correct.

Technical Reference(s): EOP 2532, Loss of Coolant Accident, rev. 037, pages26 and 29.

Provided reference(s): None

Learning Objective: 283789 ILT Outline and explain the bases for the major actions in EOP 2532, Loss of Coolant Accident.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Replacement K/A EA1.14 Secondary pressure control received 8-9-22. K/A EA1.17 was similar to Q33, and Q33 was felt to be a better question. Wrote new question. Change completed 8-10-22.

Question #4

A Loss of Coolant Accident (LOCA) has occurred inside Containment. The following are the current plant conditions:

- Pressurizer pressure is 950 psia.
- RCS Tc is 475 °F.
- Containment pressure is 4.8 psig.

Given EOP 2541, Appendix 2, Figure 2 – RCP NPSH Curve.

1. How many RCPs must be secured?

   AND

2. The RCPs must be secured within how many minutes?

A. 1. 2 RCPs
   2. 5 minutes

B. 1. 2 RCPs
   2. 10 minutes

C. 1. 4 RCPs
   2. 5 minutes

D. 1. 4 RCPs
   2. 10 minutes

Question #4 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 011 EA1.03 Importance Rating: 3.9

K/A Statement: Ability to operate and/or monitor the following as they apply to a Large-Break LOCA: RCPs

Proposed Answer: **D**

Justification: All (4) RCPs must be secured within 10 minutes. Using the given figure, Figure 2 – RCP NPSH Curve, and applying the provided Pressurizer pressure of 950 psia and RCS $T_c$ of 475 °F, the location on the curve is between the minimum pressure for RCP operation for Normal Containment and Harsh Containment. The condition provided for Containment is 4.8 psig, which is Harsh Containment. And since 950 psia and RCS $T_c$ of 475 °F is below the Harsh Containment curve all RCPs must be secured. And the Time Critical Operator Action (TCOA) associated requires all operating RCPs to be tripped no later than 10 minutes after losing adequate RCP Net Positive Suction Head (NPSH).

**K/A match.** The question requires the ability to operate and monitor (Pressurizer pressure, Containment pressure, SIAS, and RCS temperature and then secure RCPs within a credited time). And the conditions provided are for a Large-Break LOCA. There is also no overlap with the simulator scenarios that require securing RCPs because the PPC curve is used to assess NPSH not Figure 2 – RCP NPSH Curve.

Plausibility:

A. Incorrect. Tripping (2) RCPs and 5 minutes are both incorrect. This is plausible because (2) RCPs are required to be tripped when Pressurizer pressure is less than 1714 psia AND SIAS has actuated. Both of these conditions are given in the stem of the question. In addition, (2) RCPs must be secured if only one facility of power is lost because RBCCW cooling is lost. Five (5) minutes is also plausible because this was the time requirement previously for this TCOA.
B. Incorrect. Tripping (2) RCPs is incorrect. Ten (10) minutes is correct. This is plausible because (2) RCPs are required to be tripped when Pressurizer pressure is less than 1714 psia. Both of these conditions are given in the stem of the question.
C. Incorrect. Five (5) minutes is incorrect. Four (4) RCPS is correct. This is plausible because (5) minutes was the time requirement previously for this TCOA.
D. Correct.

Technical Reference(s):

1. EOP 2532, Loss of Coolant Accident (rev. 037) page 15.
2. EOP 2541, Appendix 2, Figure 2 – RCP NPSH Curve (rev. 002) page 4.
3. OP 2260, Unit 2 EOP User's Guide (rev. 017), page 43.

Provided reference(s): EOP 2541, Appendix 2, Figure 2 – RCP NPSH Curve.

Learning Objective: 294269 Trip the Reactor Coolant Pumps no later than 10 minutes after losing adequate RCP NPSH.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(13): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-26-22.

Question #5

Select how Reactor Coolant Pump (RCP) ammeters respond to the following conditions:

1. RCP amps will ___1.___ during a plant heat-up.

   AND

2. An RCP with a sheared shaft would have ___2.___ amps than the other RCPs.

A. 1. decrease
   2. lower

B. 1. decrease
   2. higher

C. 1. increase
   2. lower

D. 1. increase
   2. higher

Question #5 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 015 AA2.13 Importance Rating: 3.3

K/A Statement: Ability to determine and/or interpret the following as they apply to Reactor Coolant Pump Malfunctions: RCP Ammeter.

Proposed Answer: **A**

Justification: RCP ammeter indications will lower as the RCS temperature is raised is correct because density lowers as temperature increases. And less work is done by the RCP motor and amps lower. And a RCP with a sheared shaft would have an ammeter indication that is lower than the other RCPs is correct because the motor is only driving its own shaft, not the pump impeller. And less work is done by the RCP motor and amps lower. This question meets the K/A because the question requires the examinee to determine what RCP ammeter indications for plants conditions and a pump malfunction.

Plausibility:

A. Correct.
B. Incorrect. An RCP with a sheared shaft would have an ammeter indication that is higher is not correct. This is plausible because the examinee could confuse this with locked rotor indications or just not understand the relations between amps and conditions described.
C. Incorrect. RCP ammeter indications will higher as the RCS temperature is raised is incorrect. This is plausible because the examinee may not understand the relations between amps and RCS temperature.
D. Incorrect. RCP ammeter indications will higher as the RCS temperature is raised is incorrect. An RCP with a sheared shaft would have an ammeter indication that is higher is not correct. This is plausible because the examinee may not understand the relations between amps and RCS temperature and could confuse this with locked rotor indications or just not understand the relations between amps and conditions described.

Technical Reference(s): GFES Pump Lesson Text Pump laws.

Provided reference(s): None.

Learning Objective:

Source: New

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics

Comments: K/A replacement 12/8/21. Replaced (AA2.15) Natural circulation flow with (AA2.10) Loss of cooling or seal injection. K/A replacement 7/29/22 due to overlap with Q27. Kept Q27. New K/A AA2.13 RCP ammeter. Wrote new question. Change complete 8-10-22.

Question #6

The plant is at 100% power.

Charging Pumps are aligned as follows:

- 'CHG PP BACK-UP CNTL' switch is in the 'P1&P2' position.
- 'CHG PP OVERIDE' switch is in the 'LEV1' position

How will Pressurizer level respond to the "A" Charging Pump tripping?

Pressurizer level will stabilize ___1.___ because the ___2.___.

A. 1. at setpoint
   2. Letdown Flow Controller (HIC-110) output will lower

B. 1. below setpoint
   2. Letdown Flow Controller (HIC-110) output will lower

C. 1. at setpoint
   2. "B" Charging Pump will start

D. 1. below setpoint
   2. "B" Charging Pump will start

Question #6 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 022 AK1.03 Importance Rating: 3.6

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of Reactor Coolant Makeup: Relationship between charging flow and PZR level

Proposed Answer: **B**

Justification: When the "A" Charging pump is lost the system condition is that 44 gpm is going into the RCS and 88 gpm is coming out of the RCS (84 gpm Letdown and 4 gpm RCP CBO). The result is that Pressurizer level will lower. As Pressurizer level lowers the deviation from setpoint will send a signal to the Letdown Flow Controller (HIC-110) to lower Letdown Flow. As the deviation from setpoint becomes greater the greater the signal is to HIC-110 to throttle closed to stabilize Pressurizer level. Pressurizer level will not return to setpoint because it is a proportional controller. The examinee has to understand, without the stem stating, that the "B" Charging is in Pull-To-Lock (PTL) because at normal operating temperature and pressure the third Charging Pump is procedurally controlled and is required to be in PTL. The "CHG PP BACK-UP CNTL" switch is a three position switch. The "P1 & P2" position designates the "P1" ("A" Charging pump) as the first backup pump (first to start on a level deviation) and the "P2" ("B" Charging pump) as the second backup pump (second to start on a level deviation). This switch position also designates the lead running Charging pump as the "C" pump.

The question matches the K/A since it asks how will Pressurizer level change (knowledge of the operational implications and/or cause and effect relationships) when a Charging pump is lost (as they apply to Loss of Reactor Coolant Makeup). Which is the relationship between charging flow and PZR level.

Plausibility:

A. Incorrect. Because Pressurizer level will not stabilize at setpoint. This is plausible because many control systems restore conditions to setpoint.
B. Correct.
C. Incorrect. Because Pressurizer level will not stabilize at setpoint and the "B" Charging pump will not start. It is plausible because if the "B" Charging pump was not in PTL then it would start. The examinee must understand the normal alignment of the Charging pumps without it being specified in the stem. Procedurally the third Charging pump must be in PTL to prevent it from starting when the plant is operating.

D. Incorrect. Because the “B” Charging pump will not start. It is plausible because Pressurizer level will stabilize below setpoint and because if the “B” Charging pump was not in PTL then it would start. The examinee must understand the normal alignment of the Charging pumps without it being specified in the stem. Procedurally the third Charging pump must be in PTL to prevent it from starting when the plant is operating.

Technical Reference(s): (1) PLC-01-C, Pressurizer Level and Pressure Control System Lesson Text, rev. 7, page 30, 32, 44, & 70.
(2) OP 2304E, Charging Pumps, rev. 024, pages 2-4.

Provided reference(s): None.

Learning Objective: 281902 ILT Given the plant with a steam bubble in the pressurizer, describe the response of the Pressurizer Level and Pressure Control System when charging flow does not match letdown flow.

Source: Modified Bank #413204, 413224, 413239, & 413247 (N18Q30)

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(8): Components, capacity, and functions of emergency systems, (10) Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: NRC feedback 7-26. Feedback to put periods after the A, B, C, and D of answers. And explain in the justification what “P1” and “P2” mean as well as what is meant by the first and second Backup Charging pumps. Completed changes on 8-1-22.

Question #7

A plant startup is in progress. “A” and “B” Reactor Coolant Pumps (RCPs) are running in concurrent operation with Shutdown Cooling (SDC).

- An inadvertent Safety Injection Actuation Signal (SIAS) actuates on both facilities.

What temperature would RISE as a result of the SIAS signal?

A. SDC system return to the RCS

B. “A” and “B” RCP seals

C. Spent Fuel Pool Cooling Heat Exchanger outlet

D. Containment atmosphere

Question #7 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 026 AK3.02 Importance Rating: 3.9

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to Loss of Component Cooling Water: The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS

Proposed Answer: **C**

Justification: On a SIAS signal RBCCW to Spent Fuel Pool Cooling (SFPC) is isolated. Valves 2-RB-8.1A SFP HX (A) RBCCW Outlet and 2-RB-8.1B SFP HX (B) RBCCW Outlet close. Plausibility of the other choices is also high because there are so many components that actuate on a SIAS. The question matches the K/A. Knowledge of the actions (SFPC is isolated) as they apply to loss of Component Cooling Water (Reactor Building Closed Cooling Water) associated with the automatic actions within the CCWS resulting from the actuation of the ESFAS (SPFC valves RB-8.1A(B) close on a SIAS signal).

Plausibility:

A. Incorrect. SDC system temperature return to the RCS will not rise. This temperature would actually lower due to the RBCCW Heat Exchanger valves (SW-8.1A, B, &C) opening and providing greater cooling to RBCCW which is cooling SDC. This is plausible because the SDC Heat Exchangers are not placed in service when SIAS actuates and because many components reposition with a SIAS actuation. The SDC Heat Exchangers are placed in service upon a Sump Recirculation Actuation Signal (SRAS).
B. Incorrect. "A" and "B" RCPs seal temperature will not rise because RBCCW flow to the seals is not effected by the SIAS signal. These temperatures would actually lower due to the RBCCW Heat Exchanger valves (SW-8.1A, B, &C) opening and providing greater cooling to RBCCW. It is plausible because the CAR cooler emergency outlet valves (RB-28.3 A, B, C & D) open thereby taking some flow away from the RCPs.
C. Correct.
D. Incorrect. Containment atmospheric temperature will actually lower when the fourth CAR fan starts and the CAR cooler emergency outlet valves (RB-28.3 A, B, C & D) open. It is plausible because the examinee may understand that all the CAR cooler emergency outlet valves (RB-28.3 A, B, C & D) are normally open so this won't provide any additional cooling and the CAR fans shifting to slow provides less cooling so Containment atmospheric temperature will rise.

Technical Reference(s): (1) ESA-01-C Engineering Safety Features Actuation System Lesson Text, rev. 4/2, Tables 2 & 3, pages 53-56.
(2) RBC-00-C Reactor Building Closed Cooling Water System, rev. 9/1, pages 7, 23

Provided reference(s): None.
Learning Objective: 281949 ILT Describe the effects of the following on RBCCW System components and operating status: (A) Safety Injection Actuation Signal (SIAS)
Source: Bank #453314 Q53 NRC 2000 exam, 451419 (same question)
History:
Cognitive Level: Comprehension or Analysis
10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.
Comments: Accepted by the NRC 7-26-22.

Question #8

The plant is at 100% power.

- Pressurizer Pressure Transmitter PT-100X is selected for control.
- Pressurizer Pressure Transmitter PT-100Y is out-of-service.
- Indicated pressure on PIC-100X begins rising.
- Both Pressurizer Spray valves are opening.
- Pressure is lowering on all Pressurizer Safety channels.

What action could be used to stabilize Pressurizer Pressure?

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

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

C. Energize only the Pressurizer Backup Heaters.

D. Energize all Pressurizer Heaters (Proportional and Backup).

Question #8 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 027 AK2.10 Importance Rating: 3.7

K/A Statement: Knowledge of the relationship between a Pressurizer Pressure Control System Malfunction and the following systems or components: PZR pressure transmitters

Proposed Answer: **B**

Justification: The indications provided are indicative of a failure of the selected Pressure Transmitter, PT-100X. 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. This will allow restoration of Pressurizer pressure to 2250 psia. With the Controller in MANUAL it removes the input from the failed pressure transmitter. Lowering output raises Heater output and closes Spray valves.

Plausibility:

A. Incorrect. Raising the controller output will open the Spray valves and cause Pressurizer pressure to lower faster. This is plausible because the examinee may reason that raising the output will increase the Heater output and reduce Spray flow.
B. Correct.
C. Incorrect. Turning on the Backup Heaters manually will not maintain pressure as full Spray valve flow will be greater than the output of all Pressurizer Heaters combined (Proportional and Backup). This is plausible because the examinee could reason that additional heaters will stop the pressure reduction.
D. Incorrect. Turning on all Pressurizer Heaters (Proportional and Backup) will not restore and maintain pressure as full Spray flow will be greater than the output of <u>all</u> the Pressurizer Heaters combined (Proportional and Backup). Also, the Proportional Heaters cannot be placed in service by simply closing the breaker hand switches. Proportional Heaters operate off of the pressure controller output signal. This is plausible because the examinee could reason that additional heaters will stop the pressure reduction.

Technical Reference(s):

1. Pressurizer Level & Pressure Control System Lesson Text, PLC-01-C, rev. 7, pages 39-41.

Provided reference(s): None.

Learning Objective: 281909 – ILT for each of the following control devices associated with the Pressurizer Level & Pressure control system, explain its purpose and operation, and describe how the controlled component(s) function when the control is in a specific position and/or specific condition: h) Backup heaters and breaker control switches, i) Proportional heaters and breaker control switches, and m) Pressurizer level and pressure transmitters.

Source: ☐ Bank #416593

History: 2005 NRC exam Q9

Cognitive Level: Comprehension or Analysis

10CFR55.41(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Replaced K/A 2.12 with 2.10. NRC feedback 7-26. Feedback was to change answers "C" and "D". Make answer "C" be "Turn on only the Backup Heaters" and make answer "D" be "Turn on all Pressurizer Heaters (Proportional and Backup). Also changed justification to match answer changes. Completed changes on 8-1-22.

Question #9

An Anticipated Transient Without Scram (ATWS) occurred from 100% power.

What indicates the ATWS Mitigation Circuit Diverse SCRAM System (DSS) actuated?

A. All eight of the Trip Circuit Breakers (TCBs) are open.

B. Both of the MG Set 480 VAC supply breakers are open.

C. Both AFW pumps are running and both AFRVs are open.

D. The TCB indicating lights on C-04 are all de-energized.

Question #9 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 029 EA2.12 Importance Rating: 3.5

K/A Statement: Ability to determine and/or interpret the following as they apply to an Anticipated Transient Without Scram: Emergency feedwater (EFW)/AFW flow

Proposed Answer: **C**

Justification: The Diverse Scram System (DSS) starts the AFW pumps and opens the AFRV when both RCS pressure is greater than 2400 psia and NI Control Channel power is greater than 20%.

K/A match. The examinee must interpret the plant parameters in the answers and determine which would indicate that the ATWS Mitigation Circuit has actuated. With the actuation identified by starting AFW pumps and opening AFRVs.

Plausibility:

A. Incorrect. The Trip Circuit Breakers (TCBs) are tripped open normally by the Reactor Protection System (RPS), not by the Diverse SRAM System (DSS). This would be the "normal" indication that the Reactor tripped. It is plausible if the examinee understands that the DSS is an alternate means of shutting down the Reactor but still uses the TCBs to remove power from the CEAs.
B. Incorrect. The DSS trips open both MG Set output contactors as an additional and separate way to shut down the Reactor. The DSS does not open both of the MG Set 480 VAC supply breakers. It is plausible if the examinee remembers that MG set power is removed by the DSS, but not how it is removed.
C. Correct.
D. Incorrect. The loss of power to the TCBs busses does not cause a loss of control power to the TCBs. Loss of indicating lights would indicate a loss of Vital DC power. The removal of power from the TCBs causes the CEAs to drop into the core and would be indicated by "green" indicator lights lite. It is plausible because with normal 480 VAC Motor Control Center (MCC) breakers, a loss of power to the bus also causes a loss of power to the breaker indicating lights.

Technical Reference(s):

1. Auxiliary Feedwater Lesson Text (Power Point), AFW-00-C, VISION 55644, rev. 6/3, pages 1, 38-41
2. Engineered Safety Features Actuation System Lesson Text, ESA-01-C, rev. 4/2, pages 1, 21-23, and Figure 8, AFAS and AMSAC Interface.

Provided reference(s): None.

Learning Objective: 280924 Given a set of plant conditions/parameter values, determine if these conditions will cause AFW to auto start as given in AFW-00-C.

Source: Bank #452200
History: Used on the Millstone 2016 NRC exam, question #9.
Cognitive Level: Comprehension or Analysis
10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Comments: NRC feedback 7-26. Feedback was to remove time from the justification, since time has no bearing on the answer. Completed change on 8-1-22.

Question #10

A Steam Generator Tube Rupture (SGTR) on the #1 Steam Generator (S/G) occurred with a subsequent Loss of Offsite Power.

Current plant conditions are:

- RCS cooldown rate = 70 ºF/hr.
- Loop 1 $T_{HOT}$ = 462 ºF.
- Loop 2 $T_{HOT}$ = 450 ºF.
- RCS pressure = 600 psia.
- CETs = 450 ºF.

1. What action is required per EOP 2534, Steam Generator Tube Rupture?

   AND

2. Why is this action required?

A. 1. Raise the cooldown rate to between 80 ºF/hr and 100 ºF/hr.
   2. To ensure Shutdown Cooling is placed in service within the required time after the event.

B. 1. Lower the RCS pressure to between 470 psia and 500 psia.
   2. To minimize the volume of water leaking from the Reactor Coolant System into the affected S/G.

C. 1. Lower the cooldown rate to between 30 ºF/hr and 60 ºF/hr.
   2. To keep the loops coupled and ensure the isolated S/G is adequately cooled and depressurized.

D. 1. Raise RCS pressure to between 850 psia and 900 psia.
   2. To eliminate voiding and ensure that natural circulation flow is adequate to continue the cooldown.

Question #10 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 038 G2.4.18 Importance Rating: 3.3

K/A Statement: Steam Generator Tube Rupture: Knowledge of the specific bases for emergency and abnormal operating procedures

Proposed Answer: **C**

Justification: A difference of more than 10 ºF between loop hot leg temperatures is an indication of the Steam Generators becoming "uncoupled". As a result, the isolated Steam Generator (S/G) becomes a heat source for the Reactor Coolant System (RCS) and the cooldown of the isolated S/G begins to stall. The proceduralized method for ensuring the isolated S/G is being adequately cooled is to establish a cooldown of 30 - 60 ºF/hr (step 29 Note). The current cooldown rate of 70 ºF/hr is greater than the prescribed cooldown rate making it necessary to slow the cooldown and allow the isolated S/G to equalize with the intact S/G.

Plausibility:

A. Incorrect. Raising the cooldown rate will cool and depressurize the intact S/G, but the isolated S/G will NOT cool down and will prevent depressurizing the RCS. It is plausible because the initial direction is to perform an expeditious cooldown to a $T_h$ of less 515 ºF to isolate the affected S/G within 60 minutes. The Tech Spec limit for an RCS cooldown is 100 ºF/hr. If the examinee does not realize there is a different procedural limit on the cooldown rate, for maintaining loops coupled when in natural circulation (the loss of Offsite Power removed power from RCPs), then they may believe that the 100 ºF/hr Tech Spec limit is the only one that applies. Additionally, there is a 16-hour time requirement to place SDC is service for a SGTR. And this requires an expeditious cooldown.

B. Incorrect. Lowering RCS pressure will allow more Safety Injection flow, but will also lower subcooling below the low limit of 30 ºF (600 psia ~ 36 ºF, 500 psia ~ 17 ºF, and 470 psia ~ 11 ºF subcooled respectively). It is plausible because EOP 2534 step 10, directs the crew to maintain RCS pressure as low as possible to reduce or eliminate the primary to secondary leakage. It also directs the crew to maintain RCS pressure within the P/T limits (30 ºF subcooling). But the examinee may believe that RCS pressure should be maintained as close to saturation as possible.

C. Correct.

D. Incorrect. A head void is likely with no RCPs operating. Raising RCS pressure will help eliminate the void, but will NOT improve the cooldown on the affected S/G and will only cause the leakage from the RCS to the affected S/G to rise. This is plausible because the examinee may believe that loop differential temperature is caused by head voiding which is affecting natural circulation flow.

Technical Reference(s):

1. EOP 2534, Steam Generator Tube Rupture, rev. 030, pages 1, 16, 17, and 44.
2. EOP 2534, Steam Generator Tube Rupture Lesson Text (Power Point), E34-01-C, rev. r4c2, pages 1, 2, and 29.
3. OP 2260, Unit2 EOP User's Guide, rev. 017, pages 1, 47, and 48.

Provided reference(s): None.

Learning Objective: 283824 ILT Outline and explain the bases for the major action steps for EOP 2534, "Steam Generator Tube Rupture".

Source: Bank #451832 (453426)

History: Millstone NRC 2011 ILT exam question #11

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency procedures for the facility.

Comments: Accepted by the NRC 7-26-22.

Question #11

The plant is operating at 100% power.

An Excess Steam Demand Event (ESDE) occurs inside Containment.

1. A Main Steam Isolation (MSI) will occur at a Steam Generator (S/G) pressure of __1.__.

   AND

2. Once the faulted S/G has blown dry, the intact S/G pressure will __2.__.

A. 1. 572 psia
   2. rise

B. 1. 572 psia
   2. stabilize

C. 1. 691 psia
   2. rise

D. 1. 691 psia
   2. stabilize

Question #11 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 040 AK2.09 Importance Rating: 3.7

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Steam Line Rupture: SGS

Proposed Answer: **A**

Justification: A Main Steam Isolation (MSI) will occur at a Steam Generator (S/G) pressure of 572 psia. The intact S/G pressure will rise once the faulted S/G has blown dry. The intact S/G pressure will rise once the faulted S/G has blown dry because it is now removing RCS heat.

Plausibility:

A. Correct.
B. Incorrect. The intact S/G pressure will not stabilize once the faulted S/G has blown dry. This is plausible because the examinee could think if the faulted S/G is blown dry that RCS temperature (S/G pressure) will stop lowering (which is correct) and stabilize.
C. Incorrect: A Main Steam Isolation (MSI) will not occur at a S/G pressure of 691 psia. This is plausible because the examinee could confuse the RPS trip setpoint of 691 psia on S/G pressure. The 691 psia RPS trip prevents excessive cooldown due to a steam line break.
D. Incorrect. A Main Steam Isolation (MSI) will not occur at a S/G pressure of 691 psia. The intact S/G pressure will not stabilize once the faulted S/G has blown dry This is plausible because the examinee could confuse the RPS trip setpoint of 691 psia on S/G pressure. The 691 psia RPS trip prevents excessive cooldown due to a steam line break. And the examinee could think if the faulted S/G is blown dry that RCS temperature (S/G pressure) will stop lowering (which is correct) and stabilize.

Technical Reference(s):

1. Reactor Protection System Lesson Text, RPS-01-C, R7C2, pages 1, 11.
2. Engineering Safety Features Actuation System Lesson Text, ESA-01-C, R4C2,pages 1, 17.
3. EOP 2536 Excess Steam Demand Event, Rev. 029, pages 1, 22.

Provided reference(s): None.

Learning Objective: 283857 LOIT describe the expected plant response to an Excess Steam Demand.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: K/A replaced on 6/3/22. This was one of the ten pre-submittal questions. Replaced K/A due to concept being too similar to question 56. NRC feedback 7-26. NRC provided new K/A 7-26-22. Completed change on 8-1-22.

Question #12

The crew is aligning the Condensate pumps to feed the #1 Steam Generator (S/G) using EOP 2541, Appendix 16, LOAF Response.

Given the following plant conditions:

- S/G #1 pressure is 650 psia and slowly lowering.
- S/G #2 pressure is 420 psia and lowering.
- S/G #1 level is 293 inches and slowly lowering.
- S/G #2 level is 180 inches and lowering.
- Containment pressure is 7 psig and rising.

1. Which valves will be used to throttle feed flow?

   AND

2. What is the CONTINGENCY ACTION if a valve does NOT open?

A. 1. Feedwater Regulating Valves
   2. PUSH "DOWNCOMER RESET" reset pushbuttons.

B. 1. Feedwater Regulating Valves
   2. PUSH "MSI OVRD" pushbutton.

C. 1. Feedwater Regulating Bypass Valves
   2. PUSH "DOWNCOMER RESET" pushbuttons.

D. 1. Feedwater Regulating Bypass Valves
   2. PUSH "MSI OVRD" pushbutton.

Question #12 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 054 AA1.05 Importance Rating: 3.3

K/A Statement: Ability to operate and/or monitor the following as they apply to Loss of Main Feedwater: MFW regulating control valves

Proposed Answer: **D**

Justification: The Feedwater Regulating Bypass Valves and PUSH "MSI OVRD" pushbutton is correct. The flowpath from the Hotwell to the Steam Generator (S/G) is established with the Feedwater Regulating Bypass valves (FRBVs), step 11.1 of 2541, Appendix 16, LOAF Response. And the CONTINGENCY ACTION if the FRBVs do not open is to push the "MSI OVRD" and open the FRBVs. With the S/G pressure less than 572 psia and Containment pressure greater than 4.42 psig MSI would have actuated. In addition, the MSI on Containment pressure can't be blocked. And to gain control of the valves the MSI must be overridden.

Plausibility:

A. Incorrect. The Feedwater Regulating Valves (FRVs) are not used when establishing a flowpath from the Hotwell to the S/Gs. And the CONTINGENCY ACTION is to PUSH the "MSI OVRD" pushbutton and open FRVs since MSI has actuated. It is plausible because the FRVs would provide a flowpath and pushing the "DOWNCOMER RESET" is a CONTINGENCY ACTION for this step in the procedure.
B. Incorrect. The FRVs are not used when establishing a flowpath from the Hotwell to the S/Gs. It is plausible because the FRVs would provide a flowpath and pushing the "MSI OVRD" pushbutton is the correct CONTINGENCY ACTION for this step in the procedure.
C. Incorrect. The FRBVs is correct but the CONTINGENCY ACTION to PUSH the "DOWNCOMER RESET" pushbutton and open FRBVs is not correct since MSI has actuated. It is plausible because pushing the "DOWNCOMER RESET" pushbuttons is a CONTINGENCY ACTION for this step in the procedure.
D. Correct.

Technical Reference(s):

1. EOP 2541, Appendix 16, LOAF Response, rev. 002-00, 11-13.
2. EOP 2540D, Functional Recovery of Heat Removal, rev. 026, page 69.
3. Feedwater Control System Lesson Text, FWC-01-C, rev. 4/0, page 17.
4. Engineered Safety Features Actuation System Lesson Text, ESA-01-C, rev. 4/2, pages cover, 17.

Provided reference(s): None.

Learning Objective: 283893 LOIT Predict how operator action or inaction affects plant and system conditions concerning a loss of feedwater.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Comments: Accepted by the NRC 7-26-22.

Question #13

The plant has entered EOP 2530, Station Blackout.

- The SBO is being aligned to Bus 24E.
- Neither vital battery charger is expected to be restored.

1. When must DC Bus Load Reduction be completed?

   AND

2. What power supplies are isolated?

A. 1. 45 minutes
   2. VA10 and VA20

B. 1. 45 minutes
   2. VA30 and VA40

C. 1. 75 minutes
   2. VA10 and VA20

D. 1. 75 minutes
   2. VA30 and VA40

Question #13 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 055 EK1.05 Importance Rating: 3.8

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Station Blackout: Load shedding

Proposed Answer: **D**

Justification: DC load stripping must be completed within seventy-five (75) minutes of the loss of power (EOP 2530, step 17.b.1). Power is isolated to Vital AC instrument busses VA30 and VA40 as part of EOP 2530 (EOP 2541, Appendix 28 in steps 3 & 4).

Plausibility:

A. Incorrect. Forty-five (45) minutes and VA10/VA20 are not correct. It is plausible because within 45 minutes is the time when it is checked that any vital battery charger is expected to be restored within one hour of event initiation. VA10 and VA20 are plausible because the examinee must know which busses are stripped to extend the life of the other Vital Instrument AC busses.
B. Incorrect. Forty-five (45) minutes in not correct. VA30 and VA40 are correct. It is plausible because within 45 minutes is the time when it is checked that any vital battery charger is expected to be restored within one hour of event initiation.
C. Incorrect. VA10 and VA20 are not correct. Seventy-five (75) minutes is correct. VA10 and VA20 are plausible because the examinee must know which busses are stripped to extend the life of the other Vital Instrument AC busses.
D. Correct.

Technical Reference(s):

1. EOP 2530, Station Blackout, rev. 019, pages: cover, 3, and 16.
2. EOP 2541 Appendix 28, DC Load Reduction, rev. 003, pages: cover, 1, and 2.
3. BD-E-8, Emergency Procedure Guideline Basis Document, Station Blackout Recovery Guideline, rev. 6.0, pages: cover and 25.

Provided reference(s): None.

Learning Objective: 283760 Outline and explain the bases for the major actions in EOP 2530, "Station Blackout".

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(8): Components, capacity, and functions of emergency systems.

Comments: Accepted by the NRC 7-26-22.

Question #14

The plant tripped from 100% power due a Loss of Offsite Power.

- The Main Steam Isolation Valves (MSIVs) are closed.

Five (5) minutes post trip:

1. How will Reactor Coolant System (RCS) temperatures be responding?

   AND

2. Why are they responding in this manner?

A. 1. $T_{HOT}$ will be rising and $T_{COLD}$ will be stable.
   2. The plant is experiencing a RCS cooldown.

B. 1. $T_{HOT}$ will be rising and $T_{COLD}$ will be stable.
   2. Natural circulation heat removal is building in.

C. 1. $T_{HOT}$ will be stable and $T_{COLD}$ will be lowering.
   2. The plant is experiencing a RCS cooldown.

D. 1. $T_{HOT}$ will be stable and $T_{COLD}$ will be lowering.
   2. Natural circulation heat removal is building in.

Question #14 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 056 AK3.02 Importance Rating: 4.1

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to Loss of Offsite Power: Actions contained in AOPs

Proposed Answer: B

Justification: Reactor Coolant System (RCS) temperatures will be responding five (5) minutes post trip by $T_h$ rising and $T_c$ stable because natural circulation (NC) is building in. $T_c$ will be stable because it is being controlled by the Atmospheric Dump valve (ADV) which is in automatic with a setpoint of 920 psig. $T_h$ must rise for NC to develop and cool the core.

K/A match: Millstone's actions for a loss of offsite power are contained in an EOP and not an AOP. Millstone does not have a loss of offsite power AOP. This question tests the intent of the K/A.

Plausibility:

A. Incorrect. This is not correct because the plant will not be cooling down. This is plausible because $T_h$ will be rising and $T_c$ will be stable and because the examinee could think that a steam sneak path exists. But this steam sneak path only exists when the Condenser is available and down to 800 psia in the Steam Generators. The Condenser is not available on a loss of offsite power and the MSIVs are closed, preventing the steam sneak path.
B. Correct.
C. Incorrect. This is not correct because $T_h$ will not be stable and $T_c$ will not be lowering. This is plausible because if a NC plant cooldown was being performed then this is the indications you see five (5) minutes after a cooldown started. A steam sneak path would create these indications.
D. Incorrect. This is not correct because $T_h$ will not be stable and $T_c$ will not be lowering. This is plausible if the examinee believes that NC is established by lowering the setpoint on the ADV. This would establish NC also and would be indicated by a stable $T_h$ and a lowering $T_c$ five (5) minutes after adjusting the ADV to establish NC.

Technical Reference(s):

1. EOP 2528, Loss of Offsite Power/Loss of Forced Circulation, rev. 022, pages: cover, 10, 11, 17, and 25.
2. BD-E-7, Emergency Procedure Guideline Basis Document, Loss of Offsite Power/Loss of Forced Circulation Recovery Guideline, rev. 6.0, pages: cover 5, 8, and 21.

Provided reference(s): None.

Learning Objective: 283712 ILT Explain how the Thermal Driving Head for natural circulation is developed in the Reactor Coolant System to allow natural circulation to occur.

Source: New

History:
Cognitive Level: Comprehension or Analysis
10CFR55.41(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: NRC feedback 7-26. Feedback was to add that the plant tripped from 100% power to stem. And to put in the justification that Millstone's actions for a loss of offsite power are contained in an EOP and not an AOP, and this tests the intent of the K/A. The change was completed on 8-1-22.

Question #15

A loss of what two Vital AC Instrument buses will result in a Sump Recirculation Actuation?

A. VA10 and VA20

B. VA20 and VA30

C. VA20 and VA40

D. VA30 and VA40

Question #15 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 057 AA2.21 Importance Rating: 3.1

K/A Statement: Ability to determine and/or interpret the following as they apply to Loss of Vital AC Electrical Instrument Bus: RWST level

Proposed Answer: B

Justification: Loss of Vital Instrument Buses VA20 and VA30 will result in a Sump Recirculation Actuation (SRA). The RWST level instruments are powered from the Engineered Safety Features Actuation System (ESFAS) sensor cabinets. And each sensor cabinet has a primary power source (normal) and a second power source (alternate). Failure of both the normal and alternate power sources to a sensor cabinet will de-energize the associated RWST level indicator. Sensor cabinet "B" is powered from VA20 (normal) and has VA30 (alternate). Sensor cabinet "C" is powered from VA30 (normal) and has VA20 (alternate). With both VA20 and VA30 de-energized, RWST level channels "B" and "C" fail low and actuate a SRA.

Plausibility:

A. Incorrect: A loss of VA10 and VA20 will not result in in a SRA because the "A" and "B" sensor cabinets will not de-energize and will maintain power to the "A" and "B" RWST level instruments. Sensor cabinet "A" is still powered by its alternate source, VA40. And Sensor cabinet "B" is still powered by its alternate source, VA30. Therefore, all RWST level instruments are still powered. This is plausible because the examine must understand what powers the RWST level instruments. Most instruments that input to the ESF system are powered directly from Vital AC Electrical buses.
B. Correct.
C. Incorrect. A loss of VA20 and VA40 will not result in in a SRA because the "B" and "D" sensor cabinets will not de-energize and will maintain power to the "B" and "D" RWST level instruments. Sensor cabinet "B" is still powered by its alternate source, VA30. And Sensor cabinet "D" is still powered by its alternate source, VA10. Therefore, all RWST level instruments are still powered. This is plausible because the examine must understand what powers the RWST level instruments. Most instruments that input to the ESF system are powered directly from Vital AC Electrical buses.
D. Incorrect. A loss of VA30 and VA40 will not result in in a SRA because the "C" and "D" sensor cabinets will not de-energize and will maintain power to the "C" and "D" RWST level instruments. Sensor cabinet "C" is still powered by its alternate source, VA20. And Sensor cabinet "D" is still powered by its alternate source, VA10. Therefore, all RWST level instruments are still powered. This is plausible because the examine must understand what powers the RWST level instruments. Most instruments that input to the ESF system are powered directly from Vital AC Electrical buses.

Technical Reference(s):

1. Engineered Safety Features Actuation System Lesson Text, ESA-01-C, rev. 4/2, pages 38 & 39.
2. Engineered Safety Features Actuation System Power Point, ESA-01-C, rev. 4/2, pages 28, 91, 204.

Provided reference(s): None

Learning Objective: 281375 ILT Describe the effects on the Engineered Safety Features Actuation System (ESAS) sensor and actuation circuits, of a loss or malfunction of the following, A) Vital 120 VAC Distribution System

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-26-22.

Question #16

The plant trips from 100% power.

- Vital DC Bus DV20 is lost.

What component has lost the ability to be operated remotely?

A. Turbine Driven Auxiliary Feedwater pump

B. “A” Auxiliary Feedwater Regulating valve, 2-FW-43A

C. “B” Auxiliary Feedwater Regulating valve, 2-FW-43B

D. “B” Atmospheric Dump valve, 2-MS-190B

Question #16 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 058 AK1.06 Importance Rating: 3.7

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Loss of DC Power: Loss of remote or automatic operation.

Proposed Answer: **C**

Justification: The loss of DV20 results in the inability to remotely operate the "B" AFRV. The loss of DV20 power removes power from the DC solenoid and fails the "B" AFRV open. The AFAS will start the "A" Auxiliary Feedwater (AFW) pump and open the "A" Auxiliary Feedwater Regulating Valve (AFRV). The 'A' AFW pump a can be operated remotely, as can the Turbine Driven Auxiliary Feedwater (TDAFW). The TDAFW control power is normally aligned to DV20 but can be aligned to DV10 by a handswitch on the Main Control Boards and can therefore be operated remotely when DV20 is lost.

Plausibility:

A. Incorrect: The TDAFW (Turbine Drive Auxiliary Feedwater) pump control power is normally aligned to DV20 but can be aligned to DV10 by a handswitch on the Main Control Boards and can therefore be operated remotely when DV20 is lost. This is plausible because the examinee could think that the TDAFW pump can only be powered from DV20 and know that it can be operated locally and therefore not need to be also powered from DV10.
B. Incorrect: The "A" AFRV is not affected by the loss of DV20. It will continue to be able to be operated remotely on a loss of DV20. This is plausible because the examinee may believe the loss of DV20 fails open this valve.
C. Correct.
D. Incorrect. The "B" Atmospheric Dump valve (ADV) remote control is not affected by the loss of DV20. Control power for remote operation is from VA-20. Therefore, remote control is not affect by the loss of DV20. This is plausible because the loss of DV20 does affect the "B" ADV since it removes power to the remote position indication (red open indication light). The examinee could reason that this also affects the ability to operate this ADV.

Technical Reference(s):

1. AOP 2506B, Loss of Vital 125 VDC Instrument Panel DV20, rev. 005, pages: cover, 4, 14, & 26.
2. AOP 2504D, Loss of 120VAC Vital Instrument Panel VA-20, rev. 011, page 6.
3. AFW-00-C, Auxiliary Feedwater System Lesson Text, rev. 6, pages: cover & 14.
4. LVD-00-C, 125 VDC/120 VAC Lesson Text, rev. 7/2, pages: Cover, 51-53.

Provided reference(s): None.

Learning Objective: 282907 ILT Given a set of plant conditions, explain the mitigating actions taken for AFW per AOP 2506A or B in response to a loss of DV10 or DV20.

Source: New

History:
Cognitive Level: ☐ Comprehension or Analysis
10CFR55.41(b)(8): Components, capacity, and functions of emergency systems.
Comments: K/A replacement 12/8/21. Replaced (AK1.03) Effect of battery discharge rate on capacity with (AK1.06) Loss of remote or automatic operation. NRC feedback 7-26. Feedback was that this question was testing the K/A backwards. Question was what can be operated remotely when it should have been what remote operation capability was lost due to the loss of DV20. Wrote a new question to test the K/A correctly, changed justification, and updated references. Completed change on 8-1-22.

Question #17

The plant tripped from 100% power.

- SIAS actuated.
- Facility 1 Service Water pump tripped.

Which room will lose cooling as a result of the loss of Facility 1 Service Water?

A. East 480 VAC Switchgear room

B. 25' 6" Cable Vault

C. East DC Switchgear room

D. Upper 4160 VAC Switchgear room

Question #17 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 062 AK2.04 Importance Rating: 3.0

K/A Statement: Knowledge of the relationship between Loss of Service Water and the following systems or components: Chilled water systems

Proposed Answer: **C**

Justification: The East DC Switchgear ("A") room is cooled by Chilled Water which is cooled by Facility 1 Service Water once SIAS is actuated. And once Facility 1 Service Water is lost the East DC Switchgear room cooling is lost. They must understand that the East Room contains Facility 1 equipment.

Plausibility:

A. Incorrect. The East 480 VAC room is Facility 1 but does not have Chilled Water nor Service Water cooling it. It is plausible because the East 480 VAC room is the Facility 1 480 vital VAC room.
B. Incorrect. Cooling is lost to the 25' 6" Cable Vault due to the SIAS signal not due the loss of the Facility 1 Service Water. It is plausible because the examinee must understand that this Vital area is normally cooled by Non-Vital Chilled Water and this area is not cooled when SIAS is actuated.
C. Correct.
D. Incorrect. The Upper 4160 VAC room contains Facility 2 equipment and is cooled by Facility 2 Service Water. This is plausible because this area is supplied by Service Water. The examinee must understand what areas are cooled by which facility of Service Water.

Technical Reference(s):

1. CHW-04-C, Chilled Water System Lesson Text, rev. 8/0, pages: Cover, 3-6, 16, and Power Point drawings slides.
2. LVD-00-C, 125 VAC – 120 VAC Electrical System Lesson Text, rev. 10, Figure 10.
3. SWS-00-C, Service Water System Lesson Text, rev. 7/3, page: cover and 10.
4. P&ID 25203-26008 Service Water to Vital AC Switchgear Cooling Coil and AC Chillers, rev. 34 SH 3.

Provided reference(s): None.

Learning Objective: 282302 Describe the effects of a loss or malfunction of the Service Water System on the following: D) Vital AC Switchgear, E) DC Switchgear.

Source: ☐ New

History:

Cognitive Level: ☐ Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-26-22.

Question #18

The plant is operating at 100% power with the offsite transmission system experiencing electric grid disturbances.

- CONVEX has declared the Capacity Deficiency Alert.
- CONVEX requests maximum VARs output to support the grid.
- Generator voltage regulation is in its normal alignment.
- Main generator output = 900 MWe.
- Main generator hydrogen pressure = 60 psig.

Given AOP 2580, Degraded Voltage, Attachment 1, Estimated Capability Curves…

1. What is the VARs limit for the given conditions?

   AND

2. Which voltage regulator will be used to raise VARs?

A. 1. 360 MVARs
   2. CS-70, “MAN DC REG”

B. 1. 360 MVARs
   2. CS-90, “AUTO AC REG”

C. 1. 440 MVARs
   2. CS-70, “MAN DC REG”

D. 1. 440 MVARs
   2. CS-90, “AUTO AC REG”

Question #18 ☒ RO ☐ SRO

Tier # 1 Group # 1 K/A # 077 AA1.02 Importance Rating: 3.6

K/A Statement: Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Turbine/generator controls

Proposed Answer: **D**

Justification: The Estimated Capability Curve (Attachment 1 of AOP 2580 Degraded Voltage) for 60 psig hydrogen pressure has the limit of 440 MVARs at 900 MWe and 60 psig hydrogen pressure. This is in the Lagging direction. Lagging is sending MVARs out to the grid and Millstone 2 is not generally permitted to operate with a Leading power factor. Generator voltage adjustment are made with CS-90, "AUTO AC REG". Normal alignment for voltage regulator is the auto regulator. MVAR adjustments are made with this auto regulator.

Plausibility:

A. Incorrect. 360 MVARs is not correct nor is adjusting CS-70, "MAN DC REG". This is plausible because 360 MVARs is the maximum limit if operating with a Leading power factor. The examinee could think that Leading power factor puts MVARs out to the grid. They could also think that CS-70, "MAN DC REG" is used to adjust MVARs. And it is only used if the CS-90, "AUTO AC REG" fails.
B. Incorrect. 360 MVARs is not correct. This is plausible because 360 MVARs is the maximum limit if operating with a Leading power factor. The examinee could think that Leading power factor puts MVARs out to the grid.
C. Incorrect. Adjusting CS-70, "MAN DC REG" is not correct. This is plausible because the examinee could think that CS-70, "MAN DC REG" is used to adjust MVARs. And it is only used if the CS-90, "AUTO AC REG" fails.
D. Correct.

Technical Reference(s):

1. AOP 2580, Degraded Voltage, rev. 003-06. Pages: cover, 5 – 9 and 12 (Attachment 1)
2. OP 2204, Load Changes, rev. 046, pages: cover, 88, 89.

Provided reference(s): AOP 2580, Degraded Voltage, rev. 003-06. Attachment 1 Estimated Capability Curves (Attachment 1).

Learning Objective: 283476 ILT Given a degraded voltage condition, determine the mitigating actions required by AOP 2580, Degraded Voltage.

Source: ☐ New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: NRC feedback 7-26. Feedback was that the question should reference an electric grid disturbance in stem to tie to K/A, state in the stem that voltage regulation is in its normal alignment, and get rid of “you” in the second question. Completed change on 8-1-22.

Question #19

A reactor startup is in progress.

- The CEA Control switch is released.
- The CEDS 'OFF' pushbutton is pushed and its light is lit.

The CEAs continue to withdraw.

What actions are procedurally directed?

A. Terminate the startup and insert all Regulating Group CEAs per OP 2302A.

B. Initiate Emergency Boration in accordance with AOP 2558, Emergency Boration.

C. Suspend the reactor startup and enter AOP 2556, CEA Malfunctions.

D. Trip the reactor and enter EOP 2525, Standard Post Trip Actions.

Question #19 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 001 G2.1.20 Importance Rating: 4.6

K/A Statement: Continuous Rod Withdrawal: Ability to interpret and execute procedure steps.

Proposed Answer: **D**

Justification: With the CEA Control Switch released and the CEDS 'OFF' pushbutton illuminated, the CEAs should not be moving. The operator no longer has control of the CEAs. OP 2202, Reactor Startup requires the reactor be tripped immediately and EOP 2525 be performed (Discussion Section).

Plausibility:

A. OP 2202 Conditional Actions require the Regulating Groups be inserted if Termination Criteria are met (Attachment 5, Step 8.1). The continuous CEA withdrawal accident taking place is beyond the bounds of termination criteria.
B. OP 2202 Conditional Actions require initiation of Emergency Boration if the reactor is tripped due to an uncontrolled cooldown (Attachment 5, step 3). The operator may think the CEAs are untrippable and emergency Boration is required.
C. OP 2202 Conditional Actions direct the operator to AOP 2556 in the case of CEA Malfunctions. AOP 2556 does not address a continuous CEA withdrawal so would not provide any procedural guidance in this instance.
D. Correct

Technical Reference(s): OP 2202, Reactor Startup

Provided reference(s):none

Learning Objective:

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Accepted by the NRC 7-26-22.

Question #20

The plant was tripped.

- Multiple CEAs have not inserted.
- Reactor power is 10E-5 % and rising.
- Charging is unavailable.
- RCS pressure is 1950 psia.
- RCS $T_C$ is 515 °F.
- 2540A, Functional Recovery of Reactivity Control, has been entered.

1. What action is performed to achieve Emergency Boration?

   AND

2. How is HPSI flow initiated?

A. 1. Depressurize the RCS.
   2. Manually start a HPSI pump and ensure a HPSI injection valve is open.

B. 1. Depressurize the RCS.
   2. Allow SIAS to automatically initiate on lowering RCS pressure.

C. 1. Cooldown the RCS.
   2. Manually start a HPSI pump and ensure a HPSI injection valve is open.

D. 1. Cooldown the RCS.
   2. Allow SIAS to automatically initiate on lowering RCS pressure.

Question #20 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 024 AA1.09 Importance Rating: 3.5

K/A Statement: Ability to operate and/or monitor the following as they apply to Emergency Boration: ECCS

Proposed Answer: **A**

Justification: Depressurize the RCS without cooling down is correct. If power is rising (as is specified in the stem) then the procedure directs the cooldown be stopped and temperature be stabilized (step 1 of RC-3). Safety injection is achieved by depressurizing the RCS by controlling main or auxiliary spray or operating PORVs. HPSI injection is achieved by ensuring (starting) at least one HPSI pump is operating and ensuring (opening) at least one of the associated HPSI loop injection valves is open. SIAS is not initiated to start the HPSI pumps and open the loop injection valves. SIAS is only addressed if it has been initiated and at 1950 psia SIAS would not have been actuated.

Plausibility:

A. Correct.
B. Incorrect. Allow SIAS to automatically initiate is not correct. SIAS is only addressed if it has been initiated and at 1950 psia SIAS would not have been actuated. This is plausible because SIAS is a step in the procedure and the examinee might think that letting Safety Injection automatically initiate is appropriate to obtain HPSI injection.
C. Incorrect. Cooldown of the RCS is not correct. If power is rising (as is specified in the stem) then the procedure directs the cooldown be stopped and temperature be stabilized (step 1 of RC-3). This is plausible because you would want to cooldown the RCS to lower pressure and maintain subcooling margin, but cooldown is not correct as power is rising.
D. Incorrect. Neither Cooldown of the RCS nor allowing SIAS to automatically initiate are correct. If power is rising (as is specified in the stem) then the procedure directs the cooldown be stopped and temperature be stabilized (step 1 of RC-3). SIAS is only addressed if it has been initiated and at 1950 psia SIAS would not have been actuated. This is plausible because you would want to cooldown the RCS to lower pressure and maintain subcooling margin. Additionally, SIAS is a step in the procedure and the examinee might think that allowing Safety Injection to automatically initiate is appropriate to obtain HPSI injection.

Technical Reference(s): EOP 2540A, Functional Recovery of Reactivity Control. Rev. 010-01, pages: cover, 8, 9, and 11.

Provided reference(s): None.

Learning Objective: LOIT Describe the general approach used to recover lost of safety functions: a. EOP 2540A “Functional Recovery of Reactivity Control”.

Source: ☐ New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-26. Feedback was to remove “you” from the question stem and to check justification for answer “C”. The justification for “C” was correct but a statement was added to enhance clarity. Completed change on 8-1-22. AL comment on 8-9-22 was “open” HPSI” injection valve is not correct since the valves are already open. Changed to “ensure” a least one HPSI injection valve is open. Change completed 8-10-22.

Question #21

The crew is performing OP 2202, Reactor Startup ICCE.

What indicates a failed HIGH Wide Range Nuclear Instrument (WRNI) channel?

A. Associated Reactor Protection System (RPS) channel tripped.

B. Wide Range Nuclear Instrument (WRNI) Level 2 LED not lit.

C. All CEA motion is stopped from a CEA motion inhibit (CMI).

D. Local Power Density (LPD) and Loss of Load trips are enabled.

Question #21 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 033 AA2.02 Importance Rating: 3.3

K/A Statement: Ability to determine and/or interpret the following as they apply to Loss of Intermediate Range Nuclear Instrumentation: Intermediate range detector failure.

Proposed Answer: **C**

Justification: All CEA motion is stopped from a CEA motion inhibit (CMI). When the Wide Range Nuclear Instrument (WRNI) fails high (> $10^{-4}$ % power) the level 2 bistable trips and the "LVL 2" is lit. The level 2 bistable outputs to the CEA control system. The CEAPDS PDIL CMI is enabled when the "LVL 2" bistable is tripped at > $10^{-4}$ %. One WRNI failed high ((> $10^{-4}$ %) will cause a CEAPDS PDIL CMI and stop all CEA motion.

Plausibility:

A. Incorrect. Associated Reactor Protection System (RPS) channel tripped. There are no Reactor Protection System (RPS) trips generated from the WRNI. This is plausible because the examinee could think that the one detector feeds both the WRNI and PRNI. And if power failed high then a Variable High Power (VHP) trip would be generated when the WRNI detector failed high.

B. Incorrect. Wide Range Nuclear Instrument (WRNI) Level 2 LED not lit is not correct because a WRNI failed high (> $10^{-4}$ %) would cause the level 2 bistable to trip which would light the Level 2 LED. It is plausible because the WRNI draw level 2 does change state but it goes to "not lit" from "lit". It is reasonable to think that at less than $10^{-4}$ power, when the Zero Power Mode Bypass (ZPMB) is enable that the Level 2 LED would light indicating bypass.

C. Correct.

D. Incorrect. Local Power Density (LPD) and Loss of Load trips are enabled. A Power Range Nuclear Instruments (PRNI) failed high would result in Local Power Density (LPD) and Loss of Load trips. The Linear Power bistable (level 1) from a PRNI and Power Range Draw control the LPD and Turbine Trip inhibit. At > 15% power the LPD and Turbine trip is enabled. The WRNI detectors and PRNI detectors are separate detectors and feed into separate Nuclear Instrument Draws. Therefore, a WRNI failure will not affect the PRNIs and would not enable the LPD and Loss of Load trips. This is plausible because the examinee could think that the one detector feeds both the WRNI and PRNI.

Technical Reference(s):

1. NIS-01-C, Nuclear Instrumentation System Lesson Text, rev. 5/2, pages: Cover, 4, 5, 9, 10, 11, 19, 21, 29, and 42.
2. RPS-01-C, Reactor Protection System Lesson Text, rev. 7/2, pages: Cover and 24.
3. OP 2202, Reactor Startup ICCE, rev. 025, pages: cover, 4, and 22.

Provided reference(s): None.

Learning Objective: 281831 ILT Describe the effects of a loss or malfunction of the Nuclear Instrumentation System on the following: a) Reactor Protection System, b) Control Element Assembly Position Display System.

Source: ☐ New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: K/A replacement 12/8/21. Replaced (AA2.12) Maximum allowable channel disagreement with (AA2.02) Intermediate range detector failure. This was also one of the 10 NRC pre-submittal questions, minor edit to justification; changed justification B. plausibility to read…state but it goes to “not lit” from “lit”. Change made 6/7. Accepted by the NRC 7-28-22.

Question #22

The plant was shut down due to a Steam Generator Tube Leak.

Conditions prior to the shutdown were:

- Pressurizer pressure = 2250 psia.
- Steam Generator pressure = 850 psia.
- Primary-to-secondary leakage = 20 gpm.

Post-shutdown conditions are as follows:

- Pressurizer pressure = 1600 psia.
- Steam Generator pressure = 900 psia.

What is the approximate post-shutdown primary-to-secondary leakage?

A. 10 gpm

B. 14 gpm

C. 20 gpm

D. 28 gpm

Question #22 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 037 AK1.02 Importance Rating: 3.8

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Steam Generator Tube Leak: Leak rate versus D/P across the tube

Proposed Answer: **B**

Justification: Flow is proportional to the square root of the differential pressure. Therefore, the approximate post shutdown primary-to-secondary leakage flow is 14 gpm. The differential pressure prior to the shutdown is 1400 psia (2250 – 850 psia). The differential pressure post shutdown is 700 psia (1600 – 900 psia).

Flow 1 / Flow 2 = $\sqrt{\Delta P1}$ /$\sqrt{\Delta P2}$. Flow 2 = Flow 1 x ( $\sqrt{\Delta P2}$ /$\sqrt{\Delta P1}$ ).

Flow 2 = 20 gpm x ($\sqrt{700}$ /$\sqrt{1400}$ ) $= 14.2$ gpm.

Plausibility:

A. Incorrect. This is plausible if the examinee reasoned that half the pressure drop results in half the flow rate.
B. Correct.
C. Incorrect. This is plausible if the examinee reasoned that the leakage would not change.
D. Incorrect. This is plausible if the examinee made an algebra error in the proportion.

Technical Reference(s):

1. 193006 Fluid Static and Dynamics – Instructor Guide Power Point, rev. 3.3, pages: 1 and 12.
2. GFES questions; P2779, P2980, P3080, and P7342.

Provided reference(s): None.

Learning Objective: ILT Outline and explain the bases for the major action steps for EOP 2534, "Steam Generator Tube Rupture".

Source: Modified Bank CPNPP NRC 2011 Question 59. Changed stem pressures and wording. Also changed answers.

History: Comanche Peak Nuclear Power Plant.

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: NRC feedback 7-28. Feedback was that the last line of the justification was incorrect. The square root of 1400 was in the top and the square root of 700 was in the bottom and they should be opposite (square root of 1400 in denominator and square root of 700 in the numerator). Completed change on 8-1-22.

Question #23

The plant is operating at 100% power. Containment radiation levels rise significantly.

1. What Radiation monitors will isolate the Hydrogen Purge valves (EB-91, 92, 99, and 100)?

   AND

2. How are the vent paths configured to isolate?

A. 1. Containment Atmosphere Particulate and Gaseous
   2. One radiation monitor isolates only one vent path.

B. 1. Containment High Range
   2. One radiation monitor isolates only one vent path.

C. 1. Containment Atmosphere Particulate and Gaseous
   2. Either radiation monitor isolates both vent paths.

D. 1. Containment High Range
   2. Either radiation monitor isolates both vent paths.

Question #23 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 061 AK1.02 Importance Rating: 3.6

K/A Statement: Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to Area Radiation Monitoring System Alarms: Adverse containment conditions

Proposed Answer: **D**

Justification: The Containment High Range radiation monitors (RM 8240 and 8241) close the Hydrogen Purge valves (EB-91, 92, 99, and 100) which isolate the purge flow paths and stops the venting of Containment. Either radiation monitor will isolate both flow paths. Radiation monitor 8240 closes EB-99 which is in one flow path and EB-92 which is in the other flow path (outside ctmt valves). Radiation monitor 8241 closes EB-100 which is in one flow path and EB-91 which is in the other flow path (inside ctmt valves).

Plausibility:

A. Incorrect. The Containment Atmosphere Particulate and Gaseous radiation monitors (RM 8123A/B and RM8262A/B) do not close the Hydrogen Purge valves (EB-91, 92, 99, and 100). This is plausible because the Containment Atmosphere Particulate and Gaseous radiation monitors do close the Containment Purge valve (AC-4, 5, 6, and 7). It is reasonable to think that the Containment Atmosphere Particulate and Gaseous radiation monitors isolate venting of Containment on high radiation. It is also reasonable that one radiation monitor will isolate only one vent path (Facility dependent) since many systems operate equipment only in one train and are Facility dependent.
B. Incorrect. One Containment High Range radiation monitor isolates only one vent path (Facility dependent) is not correct. Either Containment High Range radiation monitor will isolate both flow paths. This is plausible because it is reasonable that one radiation monitor will isolate only one vent path (Facility dependent) since many systems operate equipment only in one train and are Facility dependent.
C. Incorrect. Containment Atmosphere Particulate and Gaseous radiation monitors is not correct. The Containment High Range radiation monitors (RM 8240 and 8241) close the Hydrogen Purge valves (EB-91, 92, 99, and 100) which isolate the purge flow paths and stops the venting of Containment. This is plausible because the Containment Atmosphere Particulate and Gaseous radiation monitors do close the Containment Purge valve (AC-4, 5, 6, and 7). It is reasonable to think that the Containment Atmosphere Particulate and Gaseous radiation monitors isolate venting of Containment on high radiation.
D. Correct.

Technical Reference(s):

1. Drawing 25203-26028 SH 3, P&ID Containment and Enclosure Building Ventilation, rev. 26.
2. ARP 2590A-119, HYDROGEN PURGE ISO VLVS HI RAD, rev. 000.
3. ARP 2590A -109, CH "A" CTMT AIR PARTICULATE RADIATION HI, rev. 000-02.
4. ARP 2590A -111, CH "C" CTMT AIR GASEOUS RADIATION HI, rev. 000-02.
5. OP 2314B, Containment and Enclosure Building Purge, rev. 027-00. Pages: cover, 46-48.

Provided reference(s): None

Learning Objective: 282105 ILT Describe the automatic actions and logic associated with the Containment High Range Radiation Monitoring Subsystem.

Source: ☐ New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: NRC stated SAT on 7-28-22. There was discussion of meeting the K/A relative to adverse containment conditions but in the end there was agreement that a significant radiation rise in containment (which was in the stem) met the adverse containment conditions.

Question #24

What would constitute a loss of CONTAINMENT INTEGRITY?

A. The outer Containment airlock door failed its leak test while in MODE 1.

B. The Equipment Hatch is secured by a minimum of 4 bolts in MODE 4.

C. The Containment leak rate exceeds its Technical Specification leakage value ($L_a$) while in MODE 5.

D. The Containment purge valves are open while moving irradiated fuel assemblies inside containment.

Question #24 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 069 G2.2.22 Importance Rating: 4.0

K/A Statement: Loss of Containment Integrity: Knowledge of limiting conditions for operation and safety limits.

Proposed Answer: **B**

Justification: The definition of CONTAINMENT INTEGRITY: 1.8 CONTAINMENT INTEGRITY shall exist when: 1.8.2 The equipment hatch is closed and sealed.
Having the equipment hatch closed by a minimum of 4 bolts is used in lower MODEs to allow for containment closure but is not robust enough to satisfy CONTAINMENT INTEGRITY requirements.
TS 4.6.1.1 CONTAINMENT INTEGRITY is applicable in MODEs 1-4. The LCO has a one-hour ACTION requirement to restore CONTAINMENT INTEGRITY.

Plausibility:

A. Incorrect. An air lock door failing its gasket seal test will cause that door to be inoperable but as long as the second air lock door is OPERABLE, CONTAINMENT INTEGRITY is satisfied. The examinee may think that, with one door inoperable, the airlock is inoperable and CONTAINMENT INTEGRITY not satisfied.
B. Correct.
C. Incorrect. The CONTAINMENT INTEGRITY is applicable in MODEs 1-4. Excessive leakage will prevent the operator from entering MODE 4. The examinee may think the containment is inoperable because of the excessive leakage.
D. Incorrect. The purge valves are allowed to be open during fuel movement. The purge valve are required to be closed in MODEs 1-4. The examinee may think the purge valves need also to be closed during fuel movement to prevent a dropped assembly from leaking directly to the outside atmosphere.

Technical Reference(s): Unit 2 Technical Specifications, Definitions, LCO 4.6.1.1

Provided reference(s): None.

Learning Objective: 280970 - Given any operating condition of the Primary Containment or Enclosure Building, state whether the condition requires entry into the Technical Specifications

Source: ☐ New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: NRC feedback 7-28. Feedback was to change the stem second bullet to a manual valve "required to be closed" to provide additional clarity. Completed change on 8-1-22. Issued a new K/A (K/A #069 G2.2.22 based on NRC review of question, new question written. DF 8/23/22

Question #25

The Chemistry department has recommended raising Charging and Letdown flow to the maximum limit to clean up high fission product activity in the RCS.

What describes a required action and why that action must be taken when Charging and Letdown flow is raised?

A. The second Letdown Flow Control valve must be placed in service in order to raise Letdown flow to maximum.

B. Health Physics department must be notified of changes in Letdown flow because this will change radiation levels.

C. Chemistry department must verify RCS boron concentration within six hours due to a potential change in boron from raising Letdown.

D. A second Ion Exchanger must be placed in service during maximum Letdown flow to limit the Ion Exchanger delta-pressure.

Question #25 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # 076 AK3.06 Importance Rating: 3.5

K/A Statement: Knowledge of the reasons for the following responses and/or actions as they apply to High Reactor Coolant Activity: Actions contained in EOPs or AOPs for high reactor coolant activity

Proposed Answer: **B**

Justification: Changing Letdown flow will change radiation level in the -5 penetration area because that is where the Letdown line first comes out of Containment. This is step 3.4 of AOP 2511, High Activity in RCS. This is a procedure required ALARA concern since the RCS is known to be at a higher activity level.

Plausibility:

A. Incorrect. The second Letdown valve is only placed in service if low RCS pressure precludes raising Letdown flow to the desired amount. There is no condition provided that would lower RCS pressure to the level where a second Letdown valve would be needed to raise Letdown flow. This is plausible because RCS cleanup is normally performed in MODE 5, which would require a second Letdown flow control valve be placed in service. The examinee may recall this in "normal" practice and assume it is always required.
B. Correct.
C. Incorrect. There is a requirement in OP-2204, Load Changes, that when power is going to be changed by >/= 15 % in one hour, the RCS must be sampled for IODINE within 2 – 6 hours to check for potential fuel pin leakage. This procedure also directs that Letdown flow be increased to allow for a smooth power changes, but the two requirements are not connected. This is plausible because the examinee may recall the required actions of OP-2204 and reason the two requirements are connected.
D. Incorrect. The Ion Exchangers are maintained in series, therefore placing a second Ion Exchanger in service would not affect the delta-pressure across a single Ion Exchanger. It is plausible because the examinee may think that under unusual situations, such as this, the Ion Exchangers could be aligned to improve RCS cleanup.

Technical Reference(s): AOP 2511, high Activity in RCS, rev. 000-04, pages: cover and 5.

Provided reference(s): None.

Learning Objective: 282931 ILT Outline the major actions of Control Room personnel using AOP 2511, "High Activity in RCS"

Source: ☐ Bank #452202 #

History: This was on the Millstone 2016 NRC exam (question 25) and also on the Millstone 2008 NRC exam.

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: Accepted by the NRC 7-28-22.

Question #26

The plant has entered EOP 2540, Functional Recovery.

- Containment pressure = 0.5 psig.
- Containment temperature is = 110 °F.
- Pressurizer level = 25%.
- RCS subcooling = 0 °F.
- RCS pressure = 46 psia.
- RCS $T_{HOT}$ = 275 °F.

Which parameter prevents the crew from placing Shutdown Cooling in service in accordance with the Functional Recovery procedures?

A. Pressurizer level

B. RCS subcooling

C. RCS pressure

D. RCS $T_{HOT}$

Question #26 ☒ RO ☐ SRO

Tier # 1 Group # 2 K/A # E09 EK2.13 Importance Rating: 3.5

K/A Statement: Knowledge of the relationship between Functional Recovery and the following systems or components: Shutdown cooling system

Proposed Answer: B

Justification: EOP 2540H, Long Term Actions, includes the following SDC entry conditions:

- Pressurizer level > 20%
- RCS subcooling is above the minimum operating limit of RCS P/T curve based on CETs or THOT. Subcooling must be greater than or equal to 30 °F and it is 0 °F.
- RCS pressure is less than or equal to 230 psia [190 psia].
- RCS THOT is less than 288 ℉ [277 ℉].

K/A match: This meets the K/A because it is the knowledge of the relationship between Functional Recovery (parameters that must be met) and the shutdown cooling system.

Plausibility:

A. Might think Pressurizer level needs to be higher.
B. Correct.
C. Might think RCS pressure is too low, i.e NPSH may not be met.
D. Might think RCS temperature is too high

Technical Reference(s): EOP 2540H, Long Term Actions

Provided reference(s): None

Learning Objective: 283930 Describe the philosophy of EOP 2540, with respect to:

d. Long Term Actions

Source: New

History: N//A

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: NRC feedback 7-28. Feedback was that the question seems backwards. List parameters and ask which of the given parameters would prevent entry to SDC. Also changed the RCS pressure to agree with RCS temperature for 0 °F subcooling. Completed change on 8-2-22.

Question #27

The plant is at 100% power.

- Bus 24E is aligned to Bus 24D.
- “B” RBCCW Heat Exchanger is not in service.
- The “A” RBCCW pump trips.
- Annunciator RBCCW HX TEMP HI (C-06/7, D-8) alarms.

1. Which Reactor Coolant Pumps (RCPs) are affected by the loss of the “A” RBCCW pump?

   AND

2. What actions are procedurally directed to mitigate the consequences of this malfunction?

A. 1. “A” and “C” RCPs
   2. Align and start the “B” RBCCW pump.

B. 1. “A” and “C” RCPs
   2. Trip the reactor and secure “A” and “C” RCPs.

C. 1. “B” and “D” RCPs
   2. Align and start the “B” RBCCW pump.

D. 1. “B” and “D” RCPs
   2. Trip the reactor and secure “B” and “D” RCPs.

Question #27 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 003 A2.06 Importance Rating: 3.5

K/A Statement: Ability to (a) predict the impacts of the following on the Reactor Coolant Pump System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: CCWS malfunction

Proposed Answer: **A**

Justification: The Facility 1 RBCCW header ("A" pump) supplies cooling to the "A" and "C" Reactor Coolant Pumps (RCPs). And AOP 2585, Immediate Operator Actions, directs starting the standby RBCCW pump whether it is aligned to the correct facility or not.

Plausibility:

A. Correct.
B. Incorrect. The Reactor does not need to be tripped and the "A" and "C" RCPs secured. This is plausible because tripping the Reactor and securing the affected RCPs is a Contingency Action if the standby pump can't be started. In addition, the examinee could think that since the standby pump is not aligned to the correct facility then it should not be started. The stem also includes the "RBCCW HX TEMP HI" annunciator in alarm, and this could be understood by the examinee as trip criteria, since some annunciators associated with RCPs are trip criteria.
C. Incorrect. The Facility 2 RBCCW header ("C" pump) supplies cooling to the "B" and "D" RCPs. This is plausible since the examinee must know which RCPs are supplied by which RBCCW header.
D. Incorrect. The Facility 2 RBCCW header ("C" pump) supplies cooling to the "B" and "D" RCPs. The Reactor does not need to be tripped and the "B" and "D" RCPs secured. This is plausible since the examinee must know which RCPs are supplied by which RBCCW header. Also because tripping the Reactor and securing the affected RCPs is a Contingency Action if the standby pump can't be started. It addition the examinee could think that since the standby pump is not aligned to the correct facility then it should not be started. The stem also includes the "RBCCW HX TEMP HI" annunciator in alarm, and this could be understood by the examinee as trip criteria, since some annunciators associated with RCPs are trip criteria.

Technical Reference(s):

1. AOP 2585, Immediate Operator Actions, rev. 006, pages: cover and 16.
2. AOP 2564, Loss of RBCCW, rev. 006, pages: cover, 6, 7, 10, 11, and 12.
3. OP 2330A, RBCCW System, rev. 030, pages: cover and 10.
4. ARP 2590E-048, RBCCW HX TEMP HI, rev. 000-04.

Provided reference(s): None.

Learning Objective: 286064 Given a set of plant conditions determine if any RCP trip criteria is met.
283215 ILT Given a set of plant conditions, determine the section within AOP 2564, Loss of RBCCW, that best mitigates the abnormal condition.

Source: New #

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments: Accepted by the NRC 7-28-22.

Question #28

The plant is operating at 100 % power.

- Auxiliary Steam is secured.

If either the (-) 5 or (-) 25 Auxiliary Building temperature drops below ___1.___, SP 2619A-001, Control Room Daily Surveillance, MODES 1 & 2, requires OP 2314C, Radwaste Ventilation System be referred to and ___2.___.

A. 1. 35 °F
   2. Radwaste Ventilation be removed from service

B. 1. 35 °F
   2. ensure that Boric Acid Heat Tracing is energized

C. 1. 65 °F
   2. Radwaste Ventilation be removed from service

D. 1. 65 °F
   2. ensure that Boric Acid Heat Tracing is energized

Question #28 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 004 K4.10 Importance Rating: 3.0

K/A Statement: Knowledge of Chemical and Volume Control System design features and/or interlocks that provide for the following: Minimum temperature requirements on borated systems

Proposed Answer: **C**

Justification: SP 2619A directs the operator to OP 2314C, Radwaste Ventilation, and secure ventilation if temperature drops to less than 65 degrees.
K/A match: There are no design features and/or interlocks associated with the CVCS system and the minimum temperature requirements on borated system. Millstone ensures minimum temperature requirements on CVCS borated system by monitoring temperatures and taking actions in accordance with surveillance procedures.

Plausibility:

A. 35 °F is the lower temperature limit for the RWST in MODES 3 & 4. Also 35 °F is just above freezing so the examinee could reason it is to prevent freezing in the building.
B. 35 °F is the lower temperature limit for the RWST in MODES 3 & 4. Also 35 °F is just above freezing so the examinee could reason it is to prevent freezing in the building. MP2 does have heat tracing on the piping from the Boric Acid Batch tank to the BAST and on piping associated with the RWST exposed to the weather. It also has main control board alarms associated with heat tracing and it is reasonable for a candidate to think higher boric acid piping of the CVCS would have heat tracing. Most of the majority of the boric acid piping heat tracing was removed years ago but some still exists.
C. Correct.
D. MP2 does have heat tracing on the piping from the Boric Acid Batch tank to the BAST and on piping associated with the RWST exposed to the weather. It also has main control board alarms associated with heat tracing and it is reasonable for a candidate to think higher boric acid piping of the CVCS would have heat tracing. Most of the majority of the boric acid piping heat tracing was removed years ago but some still exists.

Technical Reference(s): OP 2314C, Radwaste Ventilation.

Provided reference(s): None

Learning Objective: 282074 Identify the effects on the Radwaste Ventilation system of a loss or malfunction of the following
D) Aux Steam/Condensate Recovery

Source: New

History: N/A

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to remove “your” from the stem and tie the question to a procedure. Also add to the justification that there are no design features and/or interlocks associated with the CVCS system and the minimum temperature requirements on borated systems. And also to strengthen the plausibility for boric acid heat tracing. Completed change on 8-2-22.

Question #29

The plant experienced a Large Break LOCA 8 hours ago.

- Bus 24C is de-energized.
- EOP 2541, Appendix 18 Simultaneous Hot and Cold Leg Injection is entered.

1. What flowpath will be aligned for Simultaneous Hot and Cold Leg Injection?

   AND

2. What procedural action is required to achieve a Hot Leg flowpath?

A. 1. Primary Flowpath (Shutdown Cooling/LPSI)
   2. Align SI-651, SDC Isolation, power to Facility 2.

B. 1. Primary Flowpath (Shutdown Cooling/LPSI)
   2. Align LPSI injection valves so one valve is open to each loop.

C. 1. Alternate Flowpath (Pressurizer/HPSI)
   2. Align SI-651, SDC Isolation, power to Facility 2.

D. 1. Alternate Flowpath (Pressurizer/HPSI)
   2. Align LPSI injection valves so one valve is open to each loop.

Question #29 ☒ RO ☐ SRO

Tier #2 Group #1 K/A # 005 K2.02 Importance Rating: 3.3

K/A Statement: Residual Heat Removal System. Knowledge of electrical power supplies to the following: Containment isolation valves

Proposed Answer: **A**

Justification: If only Facility 2 power is available, then the primary flow path is utilized. This is Hot leg injection through the SDC Suction line using a LPSI pump. Shutdown Cooling suction valve, 2-SI-651 has dual power supplies. This valve is normally aligned to B-51 (480V bus powered from 4160 Bus 24C) but capable of being powered from B-61 (Facility 2).

Plausibility:

A. Correct.
B. LPSI injection valves are not required to be aligned so one valve is open to each loop. This is plausible since it is reasonable that both hot legs flowing into the top of the core would create better mixing to prevent boron stratification.
C. The alternate flowpath is not utilized in this condition. This flowpath is used when Facility 2 power is not available. This is plausible because the examinee may think a loss of Facility one power requires the alternate flowpath.
D. The alternate flowpath is not utilized in this condition. This flowpath is used when Facility 2 power is not available. This is plausible because the examinee may think a loss of Facility one power requires the alternate flowpath and reason that both hot legs flowing into the top of the core would create better mixing to prevent boron stratification.

Technical Reference(s): EOP 2541, Appendix 18, Simultaneous Hot and Cold Leg Injection

Provided reference(s): None

Learning Objective: 281310 - Select the correct flowpath available to mitigate boron precipitation following a Loss of Coolant Accident.

Source: New

History: N/A

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-28-22.

Question #30

The plant is operating at 100% power.

Select the normal electrical alignment and valve indication for “SYS ISOL, SI-651” (C-01).

| | Breaker | Disconnect | Green light |
|---|---|---|---|
| A. | Open | Open | NOT lit |
| B. | Open | Closed | NOT lit |
| C. | Closed | Open | Lit |
| D. | Closed | Closed | Lit |

Question #30 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #006 G191008 K1.11 Importance Rating: 3.3

K/A Statement: Residual Heat Removal System. Breakers, Relays, and Disconnects. Control room indication of a breaker status.

Proposed Answer: **C**

Justification: At power, the SDC Suction Isolation valves, SI-651 and SI-652, are closed with their breakers closed and disconnect switches open. This allows valve position indication without the ability to operate the valve motors.

Plausibility:

A. The examinee may think the breaker and disconnect are open, causing the light to be extinguished.
B. The examinee may think the disconnect allows power to the indicating light and opening the breaker prevents valve motion.
C. Correct.
D. The examinee my think both breaker and disconnect are closed and the valves positions' are controlled by the key being removed from the keyswitch.

Technical Reference(s): OP 2201, Plant Heatup

Provided reference(s): None

Learning Objective: 282439 – Outline the major actions for a plant heatup

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change "proper" in stem to "normal" and to ensure justification matches the correct answer. Completed change on 8-2-22.

Question #31

The plant has experienced a LOCA and is performing the actions of EOP 2532, Loss of Coolant Accident.

- SRAS has initiated and all supplemental actions performed.

EOP 2532 identifies which of the following as a cause of Containment Spray Pump motor amps becoming erratic?

A. LPSI pumps are robbing CS flow.

B. HPSI pumps are robbing CS flow.

C. Containment Sump Screens are becoming blocked.

D. Containment Spray Nozzles are becoming blocked.

Question #31 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 006 K3.03 Importance Rating: 3.8

K/A Statement: Knowledge of the effect that a loss or malfunction of the Emergency Core Cooling System will have on the following systems or system parameters: CSS

Proposed Answer: **C**

Justification: Lowering pump motor amps indicates a lowering flow. Post SRAS, CTMT Sump blockage will cause the HPSI and CS pump motors to lower or become erratic.

Plausibility:

A. Plausible if the examinee forgets that the LPSI pumps are tripped when SRAS actuates.
B. Plausible but the design of the system is for the HPSI pumps and CS pumps to run in parallel for the duration of the event.
C. Correct
D. Plausible if the examinee forgets that the CTMT Sump Screen holes are smaller than the spray nozzle openings

Technical Reference(s): EOP 2532, Loss of Coolant Accident, Step 48 'Monitor CTMT Sump Blockage'

Provided reference(s): None

Learning Objective: 283785 Describe the expected plant response to a LOCA

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: This was one of the pre-submittal questions the NRC provided feedback on June 3. Their comments were incorporated on Jun 24. The following changes were made in response to NRC comments; changed stem to reference EOP and changed stem question to read …EOP 2532, identifies which of the following as a cause of Containment Spray Pump motor amps becoming erratic. Accepted by the NRC 7-28-22.

Question #32

The plant is operating at 100% power.

Describe how a SIAS signal will affect the HPSI injection valves.

A. The valves will move to their full open position.

B. The valves will move to their throttled position.

C. The valves will not move due to their opening limit switch being made up.

D. The valves will not move due to their opening coils being removed.

Question #32 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 006 A3.06 Importance Rating: 3.7

K/A Statement: Ability to monitor automatic operation of the Emergency Core Cooling System, including: Valve lineups

Proposed Answer: **C**

Justification: The HPSI Injection valves are normally throttled open using SP 2604E.

Plausibility:

A. The examinee may confuse the HPSI valves with the LPSI valves which do throttle open.
B. The examinee may think the valves will open to a throttled position (where they are normally set) but the valves are already throttled and verified not to open any more after their position is set.
C. Correct.
D. The examinee may confuse the HPSI valves with the SIT outlet valves which have their closing coils removed. They might think removing the opening coils allows the valves to maintain their throttled position while allowing position indication

Technical Reference(s): SP 2604E HPSI System Alignment Check and Valve Tests

Provided reference(s): None

Learning Objective: 281483 – State the purpose and the operating characteristics of the following major HPSI components:
b) Injection Valves

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change "open" to "move" in answers "A" and "B" and replace "switch being met" with "switch being made up" in answer "C". Completed change on 8-2-22.

Question #33

The plant is operating at 100% power.

- A Pressurizer Code Safety Valve opens.
- Pressurizer pressure is lowering.

1. How does the Pressurizer Quench Tank pressure respond?

   AND

2. What actions are taken to mitigate this event?

A. 1. Rises until the operator recirculates and cools the Quench Tank.
   2. Perform Immediate Operator Actions then enter AOP 2585.

B. 1. Rises until the operator recirculates and cools the Quench Tank.
   2. Trip the plant then enter EOP 2525, Standard Post Trip Actions.

C. 1. Rises until the rupture disc fails, then lowers.
   2. Perform Immediate Operator Actions then enter AOP 2585.

D. 1. Rises until the rupture disc fails, then lowers.
   2. Trip the plant then enter EOP 2525, Standard Post Trip Actions.

Question #33 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #007 A2.01 Importance Rating:4.5

K/A Statement: Ability to (a) predict the impacts of the following on the Pressurizer Relief Tank/Quench Tank System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: A PORV that is stuck open, or code safety valve

Proposed Answer: D

Justification: An open Pressurizer Safety valve will discharge to the Quench Tank (QT) and raise QT pressure until the QT rupture disk fails. When the rupture disk fails (opens) QT tank pressure will lower below its peak pressure but will still be elevated above its normal pressure and will equalize at approximately Containment pressure. The reactor will be tripped and EOP 2525, Standard Post Trip Actions will be entered.

Plausibility:

A. Incorrect. Rise until the operator recirculates and cools the Quench Tank is not correct. Immediate Operator Actions (IOA) do not cover an open Pressurizer Safety valve. This is plausible because there are operator actions to take to recirculate and cool the QT on a high temperature condition (I.e., a leaky PORV or Safety). IOA is plausible because the there are IOA for closing a failed open PORV.
B. Incorrect. Rise until the operator recirculates and cools the Quench Tank is not correct. This is plausible because there are operator actions to take to recirculate and cool the QT on a high temperature condition (I.e., a leaky PORV or Safety).
C. Incorrect. The IOA do not cover an open Pressurizer Safety valve. AOP 2585, IOA is plausible because the examinee could confuse the IOA for the PORVs or reason that if there is an IOA for the PORVs then there should be one for the Pressurizer Safeties.
D. Correct.

Technical Reference(s):

1. RCS Lesson Plan RCS-01-C R10c2, pages 36, 53.

Provided reference(s): None

Learning Objective: 281978 – Describe the indications available to detect a leaking code safety valve.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: NRC feedback 7-28. Feedback was to remove “you” from the question. Completed change on 8-2-22. Changed answers A and B due to feedback from A Leone stating QT pressure would stabilize at a higher pressure. DF 8/10/22.

Question #34

OP 2310, Shutdown Cooling, limits RBCCW flow to 4800 gpm through the Shutdown Cooling Heat Exchanger to prevent…

A. damage to the heat exchanger shell.

B. erosion of the interior of the heat exchanger tubes.

C. run-out of the RBCCW pump.

D. robbing other RBCCW loads of cooling water.

Question #34 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 008 K1.08 Importance Rating: 4.3

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the Component Cooling Water System and the following systems: RHRS

Proposed Answer: **A**

Justification: RBCCW flow through the SDC Hx is limited to prevent shell side heat exchange damage. RBCCW flow is directed through the shell side of the Hx.

Plausibility:

A. Correct
B. SDC flow is also limited to 4,800 gpm to prevent tube side heat exchange damage. SDC flows through the tube side.
C. Examinee may think the extra 4800 gpm flow through the RBCCW system may exceed the pump's capacity.
D. Examinee may think that flowing 4,800 gpm through the Hx will rob cooling from other RBCCW cooled components.

Technical Reference(s):

1. OP 2310, Shutdown Cooling, Precautions.
2. Shutdown Cooling lesson text, SDC-00-C, rev.7, page 23.

Provided reference(s): None

Learning Objective:

Source: Bank #413547

History: none

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to link the stem of the question to a procedure and edit the justification of answer "B" to "tube side" not "shell side". Also, while not an NRC comment, another reference was added that specifically supported the answer and changed "erosion" in answers "A" and "B" to "damaged" since this is the wording used in the reference. Completed change on 8-2-22.

Question #35

Instrument Air is lost to 2-SW-8.1A, ‘A’ RBCCW HX TEMPERATURE CONTROL VALVE.

Describe the actions taken to control RBCCW temperature in accordance with OP 2326C, Off-Normal Service Water System Operation.

A. Take Local Manual control and throttle of 2-SW-8.1A.

B. Align Back-Up air bottles to 2-SW-8.1A.

C. Throttle 2-SW-9A, ‘A’ RBCCW HX SW OUTLET STOP.

D. Throttle 2-SW-8A, ‘A’ RBCCW HX SW INLET STOP VALVE.

Question #35 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 008 K6.14 Importance Rating: 3.3

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Component Cooling Water System: Temperature control valves for loads cooled by CCW

Proposed Answer: **C**

Justification: AOP, Loss of Instrument Air, directs the operator to OP 2326C, Off-Normal SW System Operations, if required to throttle SW to RBCCW Hx'ers. OP 2326C has the operator throttle the affected RBCCW HX SW outlet valve (SW-9A, 9B, and 9C respectively). Note that the Service Water TCVs cool the RBCCW system and control the RBCCW temperature. This in turn controls cooling to the RBCCW associated loads. The only load that is not cooled in this manner is Letdown. Letdown has its own TCV valve that throttles RBCCW flow to adjust Letdown temperature.

Plausibility:

A. The examinee might think taking manual control of the valve that had failed would be appropriate.
B. The examinee may think the air operated valve has back up air bottles in case of a failure
C. Correct
D. The examinee may think throttling the inlet valve would be correct since it can be operated from floor level.

Technical Reference(s): OP 2326C, Off-Normal SW System Operations

Provided reference(s): None

Learning Objective: 281941 – Describe the effects on the RBCCW system of a loss or malfunction of the following:
D) Instrument Air system.

Source: Bank #413269

History: none

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to add a link to a procedure in the question stem and to add information to the justification about how the cooling to RBCCW loads is controlled by Service Water TCVs. Completed change on 8-2-22.

Question #36

The plant is operating at 100% power when a plant transient results in the following:

- Pressurizer level = 69.6%.
- RCS pressure = 2315 psia.

How will the Pressurizer Pressure Control System respond?

| | Spray Valves | Backup Heaters |
|---|---|---|
| A. | Closed | De-energized |
| B. | Closed | Energized |
| C. | Partially Open | De-energized |
| D. | Partially Open | Energized |

Question #36 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 010 A3.03 Importance Rating: 3.3

K/A Statement: Ability to monitor automatic features of the Pressurizer Pressure Control System, including: PZR heater operation

Proposed Answer: D

Justification: The Pressurizer Level Control Program will energize the Pressurizer Backup heaters when Pressurizer level is greater than 3.6% above setpoint. At 69.6% Pressurizer level is 4.6% above setpoint and therefore the Backup heaters will be energized. The Pressurizer Pressure Control Program will start to open Spray valves at 2300 psia (50 psi above setpoint) and the Spray valves will be full open at 2350 psia (100 psi above setpoint). At 2315 psia the Spray valves will be partially open.

Plausibility:

A. Incorrect: The Spray valves closed and Backup heaters de-energized is not correct. This is plausible because the examinee might not know when the Spray valves open and might not know that Backup heaters energize on a level in-surge or know what level the heaters energize at.
B. Incorrect: The Spray valves closed is not correct. This is plausible because the examinee might not know when the Spray valves open.
C. Incorrect: The Backup heaters de-energized is not correct. This is plausible because the examinee might not know that Backup heaters energize on a level in-surge or know what level the heaters energize at.
D. Correct

Technical Reference(s): OP 2204, Attachment 3, Pressurizer Pressure Control Program.
OP 2204, Attachment 4, Pressurizer level Control System.

Provided reference(s): None

Learning Objective: 281901 – Given the plant with a steam bubble in the pressurizer and a given pressurizer pressure deviation from setpoint, describe the response of the Pressurizer Pressure Control system including setpoints.

Source: Modified Bank #413189

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-28-22.

Question #37

The plant is operating at 100% power.

- RPS Channel ‘A’ High Power trip is bypassed.
- RPS Channel ‘B’ power supply fails.

Based on the above conditions, what is the effect on RPS?

A. RPS will trip the unit.

B. RPS is in a 1 out of 3 logic.

C. RPS is in a 2 out of 3 logic.

D. There is no effect on RPS.

Question #37 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 012 K2.01 Importance Rating: 4.0

K/A Statement: Knowledge of electrical power supplies to the following: RPS channels, components, and interconnections

Proposed Answer: **B**

Justification: With the RPS Channel 'A' High Power trip bypassed, the Reactor Protection system is in a 2/3 logic for a reactor trip on High Power. The additional loss of RPS Channel 'B' will cause a trip signal on Channel 'B' High Power, TMLP, and LPD modules (a de-energized state equates to a tripped condition), placing RPS in a 1/3 logic for a reactor trip on High Power.

Plausibility:

A. The examinee may think that bypassing a trip puts RPS in a 1/3 logic and that the reactor will trip when the 'B' channel de-energizes.
B. Correct.
C. The examinee may recognize that RPS is in a 2/3 logic with 'A' channel bypassed but may think de-energizing the NIs has no effect on the trip logic.
D. The examinee may know that when power is lost to the RPS channel, that it trips that channel and think there is no change in the RPS logic; that it is still in a 2 of 4 logic. And can think that since a plant trip is not processed that there is no effect on RPS.

Technical Reference(s): Lesson Plan RPS-01-C Rev. 7/2 pg. 12, Logic Ladders

Provided reference(s): None

Learning Objective: 282153 – State the purpose of the following Reactor Protection System components:
6C) Logic Matrixes

Source: Modified Bank #413415

History: 2005 NRC Q#58

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change the stem question and answer "D". The stem was changed to "Based on the above conditions, what is the effect on RPS?" And answer "D" was changed to "There is no effect on RPS". Also the justification was changed for answer "D" to support the change. Completed change on 8-2-22.

Question #38

The plant is operating at 100% power.

A transient occurs with the following conditions indicated (C-04):

- MG-A Green light - LIT
- MG-A Red light - NOT LIT

These indications show the 'A' Motor Generator set ___1.___ is ___2.___ ?

A. 1. Bus 22E Feeder Breaker
   2. open

B. 1. Bus 22E Feeder Breaker
   2. closed

C. 1. Output Contactor
   2. open

D. 1. Output Contactor
   2. closed

Question #38 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 012 A4.07 Importance Rating: 3.8

K/A Statement: Ability to manually operate and/or monitor in the control room: M/G set breakers

Proposed Answer: A

Justification: The MG-A/B lights on C-04 indicate the MG set feeder breakers from Bus 22E/F and green light lit and red light not lit indicate the breaker is open.

Plausibility:

A. Correct.
B. Incorrect: The breaker lights indicate the breaker is open not closed. This is plausible because the examinee must understand what breaker light indications mean.
C. Incorrect: The Output Contactor is not correct. This is plausible because the examinee might confuse the lights with an DSS signal which opens the MG set output contactor. Light indication are correct.
D. Incorrect. The Output Contactor nor the breaker lights indicate are correct. The breaker indicating lights indicate the breaker is open not closed. This is plausible because the examinee might confuse the lights with an DSS signal which opens the MG set output contactor and examinee must understand what breaker light indications mean.

Technical Reference(s):

1. Control Element Drive System Lesson Text, CED-01-C, rev. 6 chg 5, page 6.
2. Control Element Drive System Power Point, CED-01-C, rev. 6 chg 5, page 112.

Provided reference(s): None.

Learning Objective: 281013 – Describe the purpose and basic operation of the following CEDS components and subsystems:
A) Motor Generator sets

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-28-22.

Question #39

The plant is at 100% power.

- 'D' CAR fan is in Standby.
- 2-RB-28.3D, EMERG OUTLET, is closed.
- Intake temperature is 60˚F.

An inadvertent SIAS occurs on Facility 2.

Containment temperature will ___1.___ and Facility 2 RBCCW temperature will ___2.___.

A. 1. rise
   2. rise

B. 1. lower
   2. lower

C. 1. rise
   2. lower

D. 1. lower
   2. rise

Question #39 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 013 A1.02 Importance Rating: 3.9

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the Engineered Safety Features Actuation System, including: Containment pressure, temperature, and humidity

Proposed Answer: **B**

Justification: Cooling to containment will increase. The "D" CAR fan will start in "SLOW" speed and its emergency RBCCW outlet valve will open. The "B" CAR fan will shift from "FAST" speed to "SLOW" but the overall result is additional cooling to the containment and this will lower containment temperature. Additional cooling will also be provided based on the SIAS actuation on the Service Water system. The RBCCW heat exchanger TCV summer valves will get a full open signal and provide additional cooling to the RBCCW system. There will be no changes to the "A" and "C" CAR fans.

Plausibility:

A. Incorrect. This is plausible because the examinee could reason that the "B" and "D" operating in "SLOW" would provide less cooling than one operating in "FAST" and therefore provide less cooling to containment. And reason that RBCCW would raise because more flow is going through the coolers and not understand that SIAS opened the RBCCW heat exchanger TCV summer valves.
B. Correct.
C. Incorrect. This is plausible because the examinee could reason that the "B" and "D" operating in "SLOW" would provide less cooling than one operating in "FAST" and therefore provide less cooling to containment. And that opening the emergency RBCCW outlet valve will raise the Facility 2 RBCCW temperature but not provide additional cooling because the "B" and "D" fans are in "SLOW" speed.
D. Incorrect. This is plausible because the examinee could reason that the containment temperature would lower when the emergency RBCCW outlet valve will opened on the "D" CAR fan because it provided more cooling to containment and this in turn raised Facility 2 RBCCW temperature.

Technical Reference(s):

1. Containment and Containment Systems Lesson Text, CCS-00-C, rev. 10/03, Pages: 50 & 51.

Provided reference(s): None

Learning Objective: 280980 ILT Describe the Containment Air Recirculation System design features which provide for adequate Containment cooling if a Loss of Coolant Accident or Main Steam Line Break occurs.

Source: New

History:
Cognitive Level: Comprehension or Analysis
10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Comments: Accepted by the NRC 7-28-22.

Question #40

The plant is performing a cooldown in accordance with OP 2207, Plant Cooldown.

- RCS pressure = 1600 psia and lowering.
- A Loss of Coolant Accident (LOCA) occurs inside containment.

1. Will Safety Injection (SIAS) automatically actuate at setpoint?

   AND

2. Will Containment Spray (CSAS) automatically actuate at setpoint?

A. 1. No
   2. No

B. 1. No
   2. Yes

C. 1. Yes
   2. No

D. 1. Yes
   2. Yes

Question #40 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 022K6.09 Importance Rating: 4.0

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the Containment Cooling System: ESFAS

Proposed Answer: **D**

Justification: The CTMT pressure actuation signals for SIAS, CIAS, EBFS, CSAS, MSI cannot be blocked. SIAS will actuate on CTMT pressure of 4.42 psig and CSAS will actuate at 9.48 psig with a SIAS signal present.

Plausibility:

A. Incorrect. The examinee may know that at 1600 psia in the RCS, SIAS will be blocked. OP 2207 blocks the Low RCS SIAS signal when pressure is taken below 1800 psia. They may think CSAS won't actuate without the SIAS permissive
B. Incorrect. The examinee may know that at 1600 psia in the RCS, SIAS will be blocked. OP 2207 blocks the Low RCS SIAS signal when pressure is taken below 1800 psia. Since CSAS has no block, they may assume CSAS will always actuate even without a SIAS signal
C. Incorrect. The examinee may know that SIAS will always actuate on CTMT pressure but think CSAS permissive from SIAS may get blocked when the low RCS pressure SIAS is blocked by OP 2207
D. Correct.

Technical Reference(s): ESA-00-1 Fig. 4, Engineered Safeguards Actuation Logic
OP 2208, Plant Cooldown

Provided reference(s): none

Learning Objective: 281380 – Describe each of the 11 Engineering Safety Features Actuation System (ESAS) actuations or trips with respect to:
A) Conditions sensed and setpoints

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Added RCS pressure as a Given condition based on feedback from Charles Carpenter. DF 6/16. Accepted by the NRC 7-28-22. Rewrote question based on feedback from NRC. DF 8/23/22

Question #41

What could cause runout conditions of a Containment Spray pump?

A. Clogged spray nozzles

B. Clogged Containment Sump Screens

C. Discharge line rupture

D. Suction line rupture

Question #41 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 022 G191004 K1.12 Importance Rating: 2.7

K/A Statement: Containment Cooling. Pumps. "Runout" of a centrifugal pump (definition, indications, causes, effects, and corrective measures)

Proposed Answer: **C**

Justification: Runout conditions for a centrifugal pump occur when the pump is at its maximum flow rate with minimum discharge head.

Plausibility:

A. This would present the pump running at shutoff head conditions (no flow, high head).
B. This would present the pump running at low-load conditions (low or erratic flow, low head).
C. Correct.
D. This would present the pump running at no-load conditions (no flow, low head)

Technical Reference(s): GFES – Operator Generic Fundamentals Components - Pumps

Provided reference(s):

Learning Objective: GFES – Define the following centrifugal pump associated terms: f). pump runout

Source: New

History:

Cognitive Level: Fundamental/Memory

10CFR55.41(b)(10): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change answer "B" to "Clogged Containment Sump Screens" from "closed suction valve". This was to better align with other answers. And to change cognitive level to fundamental/memory. Completed change on 8-2-22.

Question #42

The plant is operating at 100% power.

- CSAS ACTUATION SIG CH 1 TRIP (C-01, C-34), annunciates.
- P-43A, 'A' CS Pump, is operating.
- CTMT Pressure = 0.5 psig.

1. Containment Spray is designed to actuate when CTMT pressure is greater than ___1.___.

   AND

2. AOP 2571, Inadvertent ESFAS Actuation, directs what actions to secure the CS pump?

A. 1. 9.48 psig
   2. Place the pump switch in STOP, then Pull-To-Lock.

B. 1. 9.48 psig
   2. Place the pump switch in START, then STOP.

C. 1. 4.42 psig
   2. Place the pump switch in STOP, then Pull-To-Lock.

D. 1. 4.42 psig
   2. Place the pump switch in START, then STOP.

Question #42 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #026 K4.11 Importance Rating: 4.0

K/A Statement: Knowledge of the Containment Spray System design features and/or interlocks that provide for the following: Containment spray actuation signal actuation and/or reset

Proposed Answer: **B**

Justification: Containment Spray (CSAS) is designed to actuate at a CTMT pressure > 9.48 psig with an existing SIAS signal present. With the conditions given, the CSAS should NOT have occurred and the crew would enter AOP 2571, Inadvertent ESFAS Actuation. The procedure directs the operator to OVERRIDE the CSAS start signals and STOP the CS pumps (Override requires the component to be placed in its accident position then desired position)

Plausibility:

A. Correct pressure but incorrect sequence for securing the pump. The examinee may forget the CS pump does not have a PTL position.
B. Correct
C. Incorrect pressure, the examinee may remember SIAS, CIAS, EBFAS, and MSI actuate at 4.42 psig and think that CSAS will actuate along with SIAS. The examinee may forget the CS pump does not have a PTL position.
D. Incorrect pressure, the examinee may remember SIAS, CIAS, EBFAS, and MSI actuate at 4.42 psig and think that CSAS will actuate along with SIAS.

Technical Reference(s): AOP 2571, Inadvertent ESFAS Actuation.

Provided reference(s): None

Learning Objective:281141 – Describe the conditions which result in a:

1. Containment Spray Actuation Signal (CSAS)

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Rewrote question to match K/A based on NRC feedback. DF 8/10/22. Changed CTMT pressure from 15 "$H_2O$ to 0.5 psig based on discussion w/ NRC. DF 8/22/22

Question #43

The plant tripped from 100% power.

- ALL Condenser Steam Dumps are open.
- BOTH Atmospheric Dump Valves are closed.

What parameter is being used to control RCS temperature?

A. $T_{COLD}$

B. $T_{AVERAGE}$

C. S/G Pressure

D. Main Steam Line Pressure

Question #43 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #039 K5.10 Importance Rating: 3.6

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Main and Reheat Steam System: Use of $T_{ave}$ program control when steam dumping through atmospheric relief/dump valves, including $T_{ave}$ limits

Proposed Answer: **B**

Justification: With all 4 Condenser Steam Dumps open, the RCS is being controlled by TIC-4165, the T-AVE controller.

Plausibility:

A. The plant normally is maintained at 532℉ (T-COLD) post trip, the examinee may think that T-COLD is being used for controlling the Condenser Steam Dumps.
B. Correct
C. S/G Pressure controls the Atmospheric Steam Dumps, the student may confuse the Condenser steam dumps with the ADVs. Also, the ADVs receive a Quick Open signal that is generated from Tave but at this point, the ADVs are closed (no Quick Open)
D. Main Steam line pressure is used to control the 'A" Condenser Steam Dump (Turbine Bypass Valve) through PIC-4216. The examinee may think that the PIC is controlling all 4 Condenser Steam Dumps.

Technical Reference(s): Reactor Regulating System Lesson Plan, RRS-01-C (Vision #55970)

Provided reference(s): None

Learning Objective:282176 – Explain how the Reactor Regulating system calculates the following functions or parameters including the inputs used and the outputs derived from those inputs:

A. T-AVE
B. T-REF

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: MP2 does not use TAVE control for Atmospheric dumps/reliefs. Maybe K5.14 would be better, since the dumps are controlled by system pressure. Accepted by the NRC 7-28-22.

Question #44

The plant is operating at 100%.

- A Feedwater Control system transient occurs on #1 S/G.
- Steam Flow transmitter F4243A is reading normal.
- Feed Flow transmitter F5268A is reading higher than normal.
- Level transmitter L5271 is reading 3% higher than normal.
- Level transmitter L5272 is reading 3% lower than normal.

Based on the indications, what instrument is failing and how is it failing?

A. Feed Flow transmitter F5268A is failing high

B. Feed Flow transmitter F5268B is failing low

C. Level transmitter L5271 is failing high

D. Level transmitter L5272 is failing low

Question #44 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #059 G2.1.45 Importance Rating: 4.3

K/A Statement: MFW Main Feedwater System. Ability to identify and interpret diverse indications to validate the response of another indication.

Proposed Answer: **D**

Justification: Since the level indicators are diverging, one of them is failing. Level transmitters are auctioneer low so L5272 is the failing instrument. This is confirmed by the higher than average feed flow.

Plausibility:

A. A feed flow transmitter failing high would cause both level transmitters to increase since feed flow transmitter outputs are averaged then fed into the FWC circuitry. Plausible since one level transmitter is increasing.
B. A feed flow transmitter failing low would cause both level transmitters to lower since feed flow transmitter outputs are averaged then fed into the FWC circuitry. Plausible since one level transmitter is lowering. Examinee may forget S/G levels are auctioneered low.
C. Examinee may think the reason for the level divergence s that level is auctioneered high.
D. Correct.

Technical Reference(s): OP 2385, Feedwater Control System

Provided reference(s): none

Learning Objective:281450 – Given a set of plant conditions and any input failure with the FWCS in automatic mode of control:

1. Predict the response of the FWCS to the failure

Source: ☐ New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change the stem to something like, based on the indications what instrument is failing and how is it failing. Completed change on 8-2-22.

Question #45

1. State the minimum allowed operating speed for the Turbine Driven AFW pump.

   AND

2. What is the basis for this limit?

A. 1. 1400 rpm
   2. To avoid running the governor into the low speed stop.

B. 1. 1400 rpm
   2. To avoid running the pump in its resonance vibration region.

C. 1. 2000 rpm
   2. To avoid running the governor into the low speed stop.

D. 1. 2000 rpm
   2. To avoid running the pump in its resonance vibration region.

Question #45 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #061 A2.04 Importance Rating: 4.1

K/A Statement: Ability to (a) predict the impacts of the following on the Auxiliary/Emergency Feedwater System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: AFW pump failure or improper operation

Proposed Answer: **A**

Justification: From OP 2322, Auxiliary Feedwater, "To avoid operating the TDAFW governor into the low speed stop, a minimum speed of 1400RPM **must** be maintained. In some versions of Woodward governors, driving the governor into the low speed stop will cause an issue with speed control. Regardless of whether MP2 has a governor susceptible to these issues, the operator is directed to avoid the low speed stop.

Plausibility:

A. Correct
B. Correct rpm. The examinee may think the resonance region is avoided to prevent excessive vibrations.
C. 2000 rpm is the **upper** limit for warming up the TDAFW.
D. 2000 rpm is the **upper** limit for warming up the TDAFW.

Technical Reference(s): OP 2322, Auxiliary Feedwater, Precaution 3.6

Provided reference(s): None

Learning Objective: Given a set of plant conditions/parameter values, determine if a given set of plant parameters will result in a TDAFW pump trip.

Source: Modified Bank #412055

History: None

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: NRC feedback 7-28. Feedback was the question does not meet the K/A. "What is the impact of reaching the low speed stop?" Added verbiage to Justification to support question's tie to K/A -> basis for Caution in procedure to maintain speed above the low speed stop. DF 8/11/22

Question #46

The plant is operating at 100%.

- Static Switch VS-1's 'MANUAL BYPASS SWITCH' is in the 'BYPASS TO LOAD' position.

What condition would challenge the Containment design pressure if a Design Basis Accident were to occur?

A. Loss of Battery Bus 201A

B. Loss of Battery Bus 201B

C. Loss of Battery Bus 201D

D. Loss of Bus VR-11

Question #46 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #062 K5.04 Importance Rating: 3.1

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the AC Electrical Distribution System: Operation of a static inverter

Proposed Answer: **C**

Justification: With the VS-1 MANUAL BYPASS switch in the BYPASS TO LOAD position, the static switch will not swap power supplies. The BYPASS TO LOAD position indicates that Inverter 5 is supplying power to Vital AC Instrument Bus VA-10. Inverter 5 is powered from the Turbine Battery Bus 201D. CTMT pressure design is bounded by an ESD concurrent with a loss of DC power event. The Turbine battery is credited as a B/U power supply to close the FRVs (powered by VA-10/20) if the Block valves don't close (non-vital power)

Plausibility:

A. 201A is the normal supply to VA-10. The examinee may think that VA-10 is still powered from INV-1
B. The examinee recognizes that VS-1 won't swap to a good power source but may think the FRVs have dual power supplies.
C. Correct
D. The examinee may think that VA-10 power is supplied from Bus VR-11. The back-up power to VA-30 and VA-40 is VR-11&21

Technical Reference(s): OP 2345B, 120 Volt Vital Instrument AC System

Provided reference(s): None

Learning Objective: 287294 – Describe the functions of the following 120VAC system local controls, including how the components are affected by each mode or position of the control:
c) Static Switch controls (including Manual/Bypass Switch)

Source: Modified Bank #451944

History: NRC 2014 exam Q90

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: Accepted by the NRC 7-28-22.

Question #47

The plant is operating at 100% power.

The crew has entered AOP 2580, Degraded Voltage.

The procedure directs to trip the unit when the 4.16 kV buses reach ___1.___ in order to prevent safety-related pumps and motors from ___2.___.

A. 1. 2,912 V (70% nominal voltage)
   2. tripping on under-voltage

B. 1. 2,912 V (70% nominal voltage)
   2. over-heating the windings

C. 1. 3700 V (88% nominal voltage)
   2. tripping on under-voltage

D. 1. 3700 V (88% nominal voltage)
   2. over-heating the windings

Question #47 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #062 A2.16 Importance Rating: 3.7

K/A Statement: Ability to (a) predict the impacts of the following on the AC Electrical Distribution System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Degraded system voltages

Proposed Answer: **D**

Justification: AOP 2580, directs tripping the reactor at 3700 V (Level 2 UV) to protect the safety related pumps and motors from overheating. As voltage decreases, current increases as a squared function, causing overheating and potential winding damage.

Plausibility:

A. 2912V is the LNP (Level 1 UV) setpoint. T The examinee may think that preventing an undervoltage trip of equipment is why the unit is tripped.
B. 2912V is the LNP (Level 1 UV) setpoint.
C. Correct setpoint The examinee may think that preventing an undervoltage trip of equipment is why the unit is tripped.
D. Correct

Technical Reference(s): AOP 2580, Degrade Voltage

Provided reference(s): None

Learning Objective:282478 – State the plant conditions that require a reactor trip as described in AOP 2580, Degraded Voltage

Source: Modified Bank #452226,453349

History: 452226 – 2016 NRC exam Q95, 453349 – 2018 NRC exam Q18

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Question #48

The plant tripped from 100% power.

The following alarms are locked in:

- RSST LOCKOUT CHANNEL 1 (C-08, A-3).
- DIESEL GEN 13U BKR DIFFERENTIAL LOCKOUT (C-08, D-36).

Compared to an uncomplicated trip,

1. Battery 201A discharge rate is __1.__ for this trip.

   AND

2. Battery 201B discharge rate is __2.__ for this trip.

A. 1. the same
   2. the same

B. 1. the same
   2. higher

C. 1. higher
   2. the same

D. 1. higher
   2. higher

Question #48 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A # 063 K6.07 Importance Rating: 4.3

K/A Statement: Knowledge of the effect of the following plant conditions, system malfunctions, or component malfunctions on the DC Electrical Distribution System: Loss of all AC power

Proposed Answer: **B**

Justification: The RSST LOCKOUT CHANNEL 1 indicates that the RSST and North Bus are isolated and the DIESEL GEN 13U BKR DIFFERENTIAL LOCKOUT shows that the 'B' EDG is TRIPPED, not powering the 'B' vital bus. With these conditions present, the 201B Battery Charger is NOT powered while the 201A Charger IS powered. Normally, both Chargers would be powered after a trip.As such, the 201A battery discharge rate is the same as an uncomplicated trip while the 201B battery discharge rate is higher.

Plausibility:

A. Might not recognize the RSST lockout combined with the output breaker closing circuit blocked causes a loss of all AC power to Facility Z2
B. Correct
C. Might confuse nomenclature for the 'A' (12U) and 'B' (13U) EDG. Normally, Facility 2 components are even numbered while Facility 1 components are odd
D. May think both chargers are lost due to the loss of the RSST/North Bus.

Technical Reference(s): ARP 2590F-009, RSST LOCKOUT CHANNEL 1
ARP 2590F-140, DIESEL GEN 13U BKR DIFFERENTIAL LOCKOUT

Provided reference(s): None

Learning Objective:281516 – Given that a main generator trip occurred and the automatic fast transfer failed to operate, assuming no operator action, identify the loads that would be de-energized as indicated on drawing 30001

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-28-22.

Question #49

F-38A/B, Emergency Diesel Generator Room Supply Fan, starts when the associated ___1.___ and stops when the ___2.___.

A. 1. EDG speed reaches 250 rpm
   2. fan's STOP button is pushed

B. 1. EDG speed reaches 250 rpm
   2. EDG room temperature reaches 65℉

C. 1. EDG room temperature reaches 105℉
   2. fan's STOP button is pushed

D. 1. EDG room temperature reaches 105℉
   2. EDG room temperature reaches 65℉

Question #49 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #064 K1.07 Importance Rating: 3.3

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the Emergency Diesel Generators and the following systems: EDG building ventilation system

Proposed Answer: **A**

Justification: The EDG Cooling fan starts when the EDG reaches 250 rpm and needs to be secured locally by depressing the manual pushbutton.

Plausibility:

A. Correct
B. Might think the fan secures on low temperature prior the low temperature (60℉) alarm annunciating.
C. Might think the fan secures on high temperature prior the low temperature (110℉) alarm annunciating.
D. Might think the fan secures on high temperature prior the low temperature (110℉) alarm annunciating and the low temperature (60℉) alarm annunciating.

Technical Reference(s): OP 2315E, Diesel Generator Ventilation System

Provided reference(s): None

Learning Objective: NLIT State the purpose and describe the operating characteristics of the following major Emergency Diesel Generator components:
k. ventilation

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-28-22.

Question #50

RM-5099, SJAE Radiation Monitor, has its alarm setpoint adjusted weekly.

1. How is the new setpoint determined?

   AND

2. Where is the new setpoint entered?

A. 1. Chemistry will calculate and provide the new setpoint.
   2. Radiation Monitor panel RC-14 and the Plant Process Computer

B. 1. Chemistry will calculate and provide the new setpoint.
   2. Radiation Monitor panel RC-14 only

C. 1. Operations will use OP 2383C to calculate the new setpoint.
   2. Radiation Monitor panel RC-14 and the Plant Process Computer

D. 1. Operations will use OP 2383C to calculate the new setpoint.
   2. Radiation Monitor panel RC-14 only

Question #50 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #073 A4.04 Importance Rating: 3.2

K/A Statement: Process Radiation Monitoring System: Ability to manually operate and/or monitor in the control room: Alarm and/or interlock setpoint checks and adjustments

Proposed Answer: **B**

Justification: OP 2383C, Radiation Monitor Alarm Setpoint Control provides direction for making alarm setpoint changes. Under the section for adjusting setpoints on the SJAE radiation monitor, it has a step for Chemistry to provide a new setpoint. The discussion section also states that the RM-5099 setpoint is set as specified by Chemistry. Also included in OP 2383C are steps to make the setpoint change to RM-5099 (steps 4.1.5 – 4.1.12). The new setpoint is not entered in the Plant Process Computer (PPC) (step 4.1.16).

Plausibility:

A. Incorrect. Chemistry will calculate and provide the new setpoint is correct. But radiation monitor panel RC-14 and the Plant Process Computer is not correct. This is plausible because

the examinee might know that information associated with the change is inputted to the PPC but not know that the setpoint is not part of the information entered into the PPC.

B. Correct.

C. Incorrect. Operators will use OP 2383C to calculate the new setpoint is not correct. Radiation Monitor panel RC-14 and the Plant Process Computer is also not correct. This is plausible because operators, in this procedure, calculate new setpoints for the Blowdown radiation monitor RM-4262 and the RM-4262 also reads out on the PPC.

D. Incorrect. Operators will use OP 2383C to calculate the new setpoint is not correct. This is plausible because operators, in this procedure, calculate new setpoints for the Blowdown radiation monitor RM-4262.

Technical Reference(s):

1. OP 2383C, Radiation Monitor Alarm Setpoint Control, rev. 016, pages: cover, 2-11, 33, & 35.
2. SP 2833-007 SJAE Rad Monitor RM 5099 Setpoint & PPC Alarm Inputs Change Request, rev. 002-03, completed 12/28/2020.

Provided reference(s): None.

Learning Objective: 282104 ILT as contained in OP 2383C, describe the administrative controls associated with changing a setpoint for a Process or Area Radiation Monitoring System instrument channel and whether a setpoint may be changed.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change answer “D” “Operators” to “Operations” to be consistent with other answers. Completed change on 8-3-22. AL comment was that the set point is not inputted into the PPC. It is other information. So the answer identified as correct isn't. Changed stem of question to make correct. Change completed 8-10-22. NRC feedback from 8-22-22 was to change the stem second question to where is the setpoint entered. This also changed the correct answer, since the setpoint is only entered at the radiation monitor and not the PPC. Made changes to stem, answers, and justification. Change completed 8-23-22.

Question #51

The plant is operating at 100% power.

Which of the following would indicate a Service Water Heat Exchanger tube rupture?

A. Annunciator AUX BLDG SUMP LEVEL HI (C-06/7, AA-21), alarms

B. Elevated air temperature in the East 480 Volt Switchgear Room

C. Elevated air temperature in the West 480 Volt Switchgear Room

D. Elevated air temperature in the DC Switchgear Rooms

Question #51 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #076 G191006 K1.13 Importance Rating: 2.9

K/A Statement: Service Water System. Heat Exchangers and Condensers. Consequences of heat exchanger tube failure

Proposed Answer: **C**

Justification: A leak in the West DC Switchgear Room end up in the berm, activating the moisture detector, causing isolation of Service Water to heat exchanger X-181. With no heat sink, switchgear temperatures would rise.

Plausibility:

A. The examinee may think a tube leak in the RBCCW Hx would cause SW to flow into the RBCCW system (it won't: SW pressure is less than RB pressure), causing the RBCCW S/T to overflow, filling the Aux Bldg sump to the alarm setpoint..
B. Service Water does not cool the East 480V Switchgear Room, the examinee may forget this.
C. Correct
D. The non-vital chillers normally cool the DC Switchgear Rooms. Those chillers are cooled by TBCCW. The examinee may forget what cools the non-vital chiller or think the DC Rooms are cooled by the vital chillers.

Technical Reference(s): SWS-00-C, Service Water System Lesson text

Provided reference(s): None

Learning Objective: 282302 - Describe the effects of a loss of or malfunction of the Service Water system on the following:
RBCCW
TBCCW
EDGs
Vital AC switchgear
DC switchgear

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

4MAY22 – changed 'leakage' to 'rupture' based on B Bergin feedback. Mentally, leakage is conceived as maybe dripping/atomizing whereas 'rupture' is more visually impactful. NRC feedback 7-28. Feedback was to change question stem to read "which of the following" instead of "what". Completed change on 8-3-22.

Question #52

The plant was operating at 100% power.

- A loss of offsite power occurred.
- Vital bus 24C de-energized on the reactor trip and remains without power.

What is the status of the Instrument Air System?

A. NO IA compressor is supplying the IA header, “F” IA compressor available upon reset.

B. “F” IA compressor is supplying the IA header, “D” IA compressor available for backup.

C. “F” IA compressor is supplying the IA header, SA compressor available for backup.

D. “D” IA compressor is supplying the IA header, “F” IA compressor available upon reset.

Question #52 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #078 K4.04 Importance Rating: 2.9

K/A Statement: Knowledge of the Instrument Air System design features and/or interlocks that provide for the following: IAS compressor loading/unloading/starts/trips

Proposed Answer: **A**

Justification: NO IA compressor is supplying the IA header, "F" IA compressor available upon reset is correct. The "F" Instrument Air (IA) is the only compressor that has power. "F" does have power from the "B" Emergency Diesel Generator (EDG) but will not AUTO start because the bus lost power. "F" IA compressor must be reset locally in order for it to start. "D" IA compressor is powered from Non-Vital Bus 22C, which is de-energized because Bus 24C is energized from the "B" EDG. The EDG only powers vital buses. "E" IA compressor does not have power because Bus 24C is dead. This questions meets the K/A because it involves knowledge of IA design features and interlocks associated with starts and trips. Since it provides a condition and asks what IA compressors are available. Examinee must know how IACs are powered and how they respond to a condition (trip and don't auto start without local operator action).

Plausibility:

A. Correct.
B. Not correct. "F" IA compressor is not supplying the IA header because it must be reset after it losses power to restart it. "D" IA compressor is not available for backup because it's Non-Vital bus does not have power. This is plausible because on a normal plant trip this is how the IA system would function.
C. Not correct. "F" IA compressor is not supplying the IA header because it must be reset after it losses power to restart it. SA IA compressor is not available for backup because it's Non-Vital bus does not have power. This is plausible because on a normal plant trip this is how the IA system would function.
D. Not correct. "D" IA compressor is not supplying the IA header because it's Non-Vital bus does not have power. "F" IA compressor is available if reset. This plausible because on a normal plant trip this is how the IA system would function.

Technical Reference(s):

1. Instrument Air Lesson Plan IAS-00-C, rev 9/chg 5, pages 18 and 19.
2. OP 2332A, Station Air System, rev. 031-00, page 4.
3. EOP 2541, Appendix 4 , rev. 009, page 6.

Provided reference(s): None

Learning Objective: 281598 ILT Identify the effects on the Station and Instrument Air Systems of a loss or malfunction of the following: A) 4160 VAC Vital and Non-Vital Electrical Distribution Systems.

Source: Direct 451767

History: NRC 2008 Q53

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was a new K/A must be selected because the there are no design features and/or interlocks associated with isolation if instrument air to containment. A new K/A was provided on 7-29-22. The K/A provided was 078 K4.04. A new question written 8-8-22. Changed answer from “B” to “A” for the count.

Question #53

The plant is operating at 100% power. A low pressure weather system is moving across Connecticut.

1. Indicated Containment pressure will ___1.___ as the system moves closer to the site.

   AND

2. What actions are taken, if the indicated Containment pressure approaches Administrative limits?

A. 1. increase
   2. Lower pressure using Hydrogen Purge and EBFS systems.

B. 1. increase
   2. Lower pressure using only the Hydrogen Purge system.

C. 1. decrease
   2. Raise pressure using Hydrogen Purge and EBFS systems.

D. 1. decrease
   2. Raise pressure using only the Hydrogen Purge system.

Question #53 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #103 A2.06 Importance Rating: 4.5

K/A Statement: Ability to (a) predict the impacts of the following on the Containment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: High containment pressure

Proposed Answer: **A**

Justification: A low pressure weather system will cause indicated containment pressure to read high (the pressure gages are dP cells). OP 2314B, Containment and Enclosure Building Purge, directs the operator vent containment using the hydrogen purge flow path and EBFS to the Millstone stack on a high pressure condition in CTMT

Plausibility:

A. Correct.
B. Opening the Hydrogen Purge valves only would lower containment pressure but would expose the enclosure building to potentially radioactive contaminants that are inside containment.
C. Containment pressure would not decrease. The operator may think the low pressure system causes CTMT pressure to decrease. Might also think using EBFS would be a good way to raise pressure.
D. Containment pressure would not decrease. The operator may think the low pressure system causes CTMT pressure to decrease. The actions are correct for a negative containment pressure indication.

Technical Reference(s): OP 2314B, Containment and Enclosure Building Purge, Section 4.13, Venting Containment using Hydrogen Purge flow path and EBFS to Millstone Stack

Provided reference(s): None

Learning Objective: 280970 Given any operation condition for the Primary Containment or Enclosure Building, state whether the condition required entry into the Technical Specifications

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: Rewrote question to more closely match K/A. Previous question focused on temperature vice pressure. DF 23MAY22. NRC feedback 7-28. Feedback was to change stem of the question to remove “you” and add in “indicated” Containment pressure. Completed change on 8-3-22.

Question #54

The plant trips from 100% power.

- $T_{COLD}$ = 525 °F.

Given this $T_{COLD}$, which of the following would cause this condition?

A. Loop 2 $T_{HOT}$ control channel temperature failing to 515 °F.

B. Main Condenser vacuum at 10 “Hg back pressure.

C. Failure of electrical buses to transfer to the RSST.

D. TIC-4165, Steam Dump $T_{AVE}$ Controller, in manual and closed.

Question #54 ☒ RO ☐ SRO

Tier # 2 Group # 1 K/A #039 K3.06 Importance Rating: 3.3

K/A Statement: Knowledge of the effect that a loss or malfunction of the Main and Reheat Steam System will have on the following systems or system parameters: SDS.

Proposed Answer: **C**

Justification: A failure of Electrical Bus 24B to transfer to the RSST will create a sneak path of steam that will lower Steam Generator (SG) pressure. This will result in RCS temperature lowering to ~518 °F (518 °F is the saturation pressure for 800 psia). EOP 2525, Standard Post Trip Actions requires the MSIVs to be closed when SG pressure lowers to 800 psia. The loss of 24B removes power from MS-2A and MS-85A which prevents the MOVs from closing on the reactor trip. These open valves result in Main Steam being directed through the MSR second stage reheat tube bundle and directly to the Condenser.

Plausibility:

A. Incorrect. Loop 2 $T_H$ temperature failed low at 515 °F is not correct. This failure only effects the area demand signal and would require a higher actual RCS temperature to open the CDVs on an area demand signal. The Turbine Bypass valve will still function on pressure control and maintain RCS temperature. This is plausible because this would raise the temperature that would open the CDVs with the area demand signal.
B. Incorrect. Main Condenser vacuum at 10 inches Hg back pressure is not correct. The interlock on Condenser vacuum, that prevents the CDVs from opening, is greater than or equal to 15 in Hg back pressure. Therefore, vacuum will not prevent the CDVs from opening. This is plausible since the examinee may not know the value of the vacuum interlock and think that being on the Atmospheric Dump valves would put temperature outside the normal post trip band.
C. Correct.
D. Incorrect. Temperature Indicating Controller TIC-4165 in manual closed is not correct. This will prevent the area demand signal from opening the CDVs. The Turbine Bypass valve will still function on pressure control and maintain RCS temperature. This is plausible since the examinee may believe all signals are blocked to the CDVs.

Technical Reference(s):

1. Moisture Separator / Reheaters Lesson Power Point, rev. 7/2, pages: 1 & 59.
2. Reactor Regulating System Lesson Power Point, rev. 5/4, pages 1, 19 & 30.
3. EOP 2525, Standard Post Trip Actions, rev. 029, pages: cover &15.
4. AOP 2502B, Loss of Non-Vital 4.16 KV Bus 24B. rev. 005-00, pages 1, 22 & 23

Provided reference(s): None.

Learning Objective: 283651 ILT Given a set of plant conditions, predict how operator action or inaction affects plant and system conditions concerning SPTAs.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change stem of the question give a low out of band Tcold but don't tell them it is low out of band. And reword question…Given a Tcold of 525 °F which of the following would cause this? Completed change on 8-3-22.

Question #55

What Reactor Protection System trip prevents a Departure from Nucleate Boiling?

A. Power Level - High

B. Reactor Coolant Flow - Low

C. Steam Generator Water Level - Low

D. Local Power Density - High

Question #55 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #002 K3.02 Importance Rating: 4.3

K/A Statement: Knowledge of the effect that a loss or malfunction of the Reactor Coolant System will have on the following systems or system parameters: Fuel

Proposed Answer: **B**

Justification: The Low RCS flow basis provides core protection from having a Departure from Nucleate Boiling Ratio < 1.17 (protects the nuclear fuel from overheating)

Plausibility:

A. The high power trip protects the core from reactivity excursions which are too fast for the high pressure or TM/LP trips. The examinee may think the increased neutron flux will cause higher fuel temperatures and boiling to occur
B. Correct
C. The Low S/G level trip is based on maintaining the RCS heat sink and protecting the RCS from Over-pressurization. The examinee may think a loss of the heat sink could cause the RCS to boil
D. The Local Power Density trip protects against exceeding Peak Centerline Temperature. The examinee may think a loss of cooling flow will cause fuel temperature to rise causing the RCS to boil

Technical Reference(s): MP2 T.S. Bases 2.2. LIMITING SAFETY SYSTEM SETTINGS

Provided reference(s):

Learning Objective: 282159 – For each Reactor Protection System reactor trip, describe the Technical Specifications Basis and required trip setting.

Source: Modified Bank #486584

History: 2020 NRC Exam Q #39

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: Accepted by the NRC 7-28-22.

Question #56

The plant is at 70%, raising power to 100%.

- Annunciator RX POWER ΔT CH DEVIATION (C-04, AA-8), alarms.

This alarm annunciated due to the NI instruments reading ___1.___ than the ΔT power indications.

Increasing ___2.___ causes the NIs to de-calibrate.

A. 1. higher
   2. RCS temperature

B. 1. higher
   2. neutron flux

C. 1. lower
   2. RCS temperature

D. 1. lower
   2. neutron flux

Question #56 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #015 G2.1.7 Importance Rating: 4.4

K/A Statement: Nuclear Instrumentation System. Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation

Proposed Answer: A

Justification: As reactor power is increased, RCS temperature rises, causing its density to lower, allowing more neutrons to reach the NI detectors, causing higher readings. The RCS ΔT instrumentation does not decalibrate with changes in RCS density.

Plausibility:

A. Correct.
B. Incorrect. Higher is correct but neutron flux is not correct. This is plausible because the examinee may think changes in neutron flux cause the NIs to de-calibrate.
C. Incorrect. Lower is not correct but RCS temperature is correct. The examinee may know that lower temperatures allow for fewer thermal neutrons to reach the NIs but forget that a power ascension raises RCS temperature.
D. Incorrect. Lower nor neutron flux are correct. The examinee may think the nuclear instruments de-calibrate as power rises, which they do, but de-calibrate in the opposite direction such that they read lower as neutron flux increases.

Technical Reference(s): OP 2204, Load Changes
AOP 2590C-057, RX POWER ΔT CH DEVIATION (C-04, AA-8)

Provided reference(s): None

Learning Objective: 282562 – Evaluate control board indications as specified in OP 2204, Load Changes

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: One of the 10 NRC pre-submittal questions, minor edit to justification. NRC's comment was to change justification for either answers "C" or "D". One should not be "fewer" and the other "more". Change made 6/9 by John. Another edit occurred 6/16. DF. NRC feedback 7-28. Feedback was to fix the plausibility for answer "D", specifically to remove the reference to temperature. Completed change on 8-3-22.

Question #57

The plant is operating at 100% power.

- $T_{AVERAGE}$ = 569 °F.
- $T_{HOT}$ = 593 °F.
- $T_{COLD}$ = 545 °F.

Subsequently, Loop 2 $T_{HOT}$ fails to 533 °F.

1. What $T_{AVERAGE}$ will be indicated?

   AND

2. How will the failed instrument be removed from the calculation?

A. 1. 554 °F
   2. Automatically removed by Foxboro.

B. 1. 554 °F
   2. Manually removed by the Operator.

C. 1. 569 °F
   2. Automatically removed by Foxboro.

D. 1. 569 °F
   2. Manually removed by the Operator.

Question #57 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #016 A4.03 Importance Rating: 3.4

K/A Statement: Ability to manually operate and/or monitor Non-Nuclear Instrumentation in the control room: Removing a failed channel from the circuit logic

Proposed Answer**:** B

Justification: The $T_{HOT}$ instrument failed but it is within the -2% to +102% of the normal range (515 – 615 °F). Therefore, Foxboro will NOT automatically deselect the failed channel. The Operator will manually remove the failed instrument using the Foxboro system. The calculated $T_{AVERAGE}$ is now 554 °F; [(593 + 533 + 545 +545) / 4 = 554 °F

This question meets the K/A because the RRS system is a Non-Nuclear system that is monitored and controlled in the Control Room and concerns a failed channel and when it is removed from the logic circuit.

Plausibility:

A. Incorrect. The calculated Tavg is correct. However the failed instrument with not be automatically remove from the calculation by Foxboro because it is not outside the -2% to +102% range (513 – 617 °F). This is plausible because the examinee could reason that Foxboro would compare temperatures and automatically remove failed instruments based on a smaller deviation.
B. Correct.
C. Incorrect. Tavg will not be 569 °F and Foxboro IA will not automatically deselect the failed channel. This is plausible if the examinee does not understand when the RRS will remove a failed temperature element.
D. Incorrect. Tavg will not be 569 °F. This is plausible because the examinee may reason that a substitute value (good $T_{HOT}$) is inputted and that is why the 100% Tavg value did not change but that the failed $T_{HOT}$ must still be manually removed.

Technical Reference(s):

1. RRS-00-C, Reactor Regulating System Text, rev. r5c3, pages 26, 27.
2. OP 2204, Load Changes, rev. 047, pages 63, 64.

Provided reference(s): None

Learning Objective: 282175 – Describe the response of the Reactor Regulating system to a failed RCS temperature or steam pressure input during steady state and transient conditions.

Source: Modified #451868

History: NRC Exam 2011 Q57

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change the stem second question to how will the failed instrument be removed from the calculation. And change the second part of each answer to automatically removed or manually removed. Also changed the justifications to match the changes in the answers. Completed change on 8-3-22.

Question #58

1. Where are Inadequate Core Cooling (ICC) system temperatures sensed?

   AND

2. What combination of RCS pressure and temperature is used to determine Subcooling Margin?

A. 1. Above the core region
   2. Highest pressure and lowest temperature

B. 1. Above the core region
   2. Lowest pressure and highest temperature

C. 1. In the core region
   2. Highest pressure and lowest temperature

D. 1. In the core region
   2. Lowest pressure and highest temperature

Question #58 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #017 K4.01 Importance Rating: 3.9
K/A Statement: Knowledge of In-Core Temperature Monitor System design features and/or interlocks that provide for the following: Input to subcooling monitors

Proposed Answer: **B**
Justification: The temperature indication for the Inadequate Core Cooling (ICC) system are generated from the highest of the two available THOT and TCOLD signals, the second highest CET (CET High), the highest CET (CET Max), and the highest of the top three Unheated Sensors (UHTC Max). All these temperatures are above the core region (above the fuel). The input to the subcooling monitor uses the lowest pressure combined with the highest temperature to determine the minimum subcooled margin.
Plausibility:

A. Incorrect. Above the core region (above the fuel assemblies) is correct. The highest pressure and lowest temperature is plausible because the examinee may not understand the concept of subcooling and what would provide the least subcooled margin.
B. Correct.
C. Incorrect. In the core region is not correct. Highest pressure and lowest temperature is not correct. This is plausible because the examinee may think that temperature is monitored in the core region (around the fuel assemblies). A common misconception is that the Reactor Vessel Level Monitoring System (RVLMS) measures level in the core region when it actually monitors level and temperatures from just above the core to the top of the reactor vessel head. The highest pressure and lowest temperature is plausible because the examinee may not understand the concept of subcooling and what would provide the least subcooled margin.
D. Incorrect. In the core region is not correct. This is plausible because the examinee may think that temperature is monitored in the core region (around the fuel assemblies). A common misconception is that the RVLMS measures level in the core region when it actually monitors level and temperatures from just above the core to the top of the reactor vessel head.

Technical Reference(s): OP 2387G, Inadequate Core Cooling System
Provided reference(s): None
Learning Objective: 281500 – State the purpose of the Inadequate Core Cooling System
Source: New
History:
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change the first part of the answers from “fuel assemblies” to “in the core region” to create a better distractor and more accurately reflect the region monitored. Also changed the justifications to match the changes in the answers. Completed change on 8-3-22.

Question #59

A plant power ascension is in progress.

- Nuclear power and Delta-T power are BOTH at 50%.

Given these conditions, what is the programmed value for Pressurizer Level?

A. 47 - 48%

B. 50 - 51%

C. 53 - 54%

D. 56 - 57%

Question #59 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #011 K1.02 Importance Rating: 3.1

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the Pressurizer Level Control System and the following systems: RCS

Proposed Answer: **C**

Justification: OP-2204, Load Changes, Attachment 8 shows the expected value for Tavg at 50% power as 550 °F. Using this value for Tavg and Attachment 5 gives a Pressurizer Level of 53%. The examinee determines the expected Pressurizer level for 50% power, without the 2204 attachments, by knowing the Pressurizer Level program starts at 40% level, then starts to ramp up linearly at Tavg = 537.5 °F (15% power) until it reaches 65% level, when Tavg = 561.5 °F (80% power). Then a ratio can be used where level rises 65% - 40% = 25% over the power range of 80% - 15% = 65%. Power rise 15% to 50% = 35% over a range of 65%. And Pressurizer level rises X% over the range of 25%. The ratio of 35% power/65% power = X% PZR level at 50%/25% PZR level range. (35)(25)/(65) = X% change from 40% level. X% = 13.5%. The 50% power PZR level is then obtained by adding the 13.5% level change to the 40% PZR level to find the expected PZR level at 50% power of 53.5%.

Plausibility:

A. Incorrect. This is plausible because of the information the examinee must know from memory and be able to use this information to calculate the expected Pressurizer level.
B. Incorrect. This is plausible because of the information the examinee must know from memory and be able to use this information to calculate the expected Pressurizer level. Also the student could think that simply that 50% power would be 50% Pressurizer level.
C. Correct.
D. Incorrect. This is plausible because of the information the examinee must know from memory and be able to use this information to calculate the expected Pressurizer level.

Technical Reference:

1. OP 2204, Load Changes, Attachment 5, Pressurizer Level Setpoint Program, and Attachment 8, Calorimetric Inputs and Plant Parameters at Various Power Levels, rev. 047.
2. PLC-01-C, Pressurizer Level & Pressure Control System, Figure 1 and Figure 2, rev. 7.

Provided reference: None

Learning Objective: 281906 – Calculate the pressurizer level program setpoint expected for a given reactor power level or RCS temperature iaw OP 2204, Load Changes

Source: Modified Bank #449614

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to change stem to make nuclear and delta power a bullet and ask the questions as "given these conditions what is the programmed level for Pressurizer Level?". And correct typo (505 power, should be 50%). Completed change on 8-3-22.

Question #60

The Unit is moving fuel.

- "A" LPSI is operating on Shutdown Cooling (SDC).
- Bus 24C is being supplied from the RSST.
- A fuel bundle is being lifted from its core location.

The RSST is lost and the "A" Emergency Diesel Generator re-energizes Bus 24C.

What actions are required by AOP 2572, Loss of Shutdown Cooling for the following?

1. Refueling will be ___1___.

   AND

2. The ___2___ pump will be used to restore SDC.

A. 1. stopped without lowering the bundle into the core
   2. "A" LPSI

B. 1. stopped without lowering the bundle into the core
   2. "A" Containment Spray

C. 1. stopped after lowering the bundle back into its core location
   2. "A" LPSI

D. 1. stopped after lowering the bundle back into its core location
   2. "A" Containment Spray

Question #60 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #034 A2.06 Importance Rating: 3.7

K/A Statement: Ability to (a) predict the impacts of the following on Fuel Handling Equipment System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Loss of RHR cooling flow

Proposed Answer: **C**

Justification: Refueling will be stopped after lowering the bundle back into its core location is correct. The "A" LPSI pump will be used to restore SDC is correct. The fuel bundle will be lowered back into the core to place the fuel in a safe location. OP 2209A, Refueling Operations specifies in step 2.4.1 that suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. The "A" LPSI pump will be used because a Containment Spray pump is only used if a LPSI pump can not be used. In this case the "A" LPSI pump is available so it should be used. K/A justification: The RHR system for Millstone Unit 2 uses the Low Pressure Safety Injection (LPSI) system. LPSI pumps are the RHR pumps. Predict the impact on Fuel Handling Equipment System is stopping fuel movement and specifically how it will be stopped. And based on those predictions, use procedures to correct, control, or mitigate the consequences of the loss of RHR cooling flow is starting the LPSI pump.

Plausibility:

A. Incorrect. Stopped without lowering the bundle into the core is not correct. This is plausible because the Loss of Shutdown Cooling procedure directs stopping CORE ALTERATIONS. The examinee may not understand that the refueling procedure and Technical specifications state that suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
B. Incorrect. Stopped without lowering the bundle into the core is not correct. The "A" Containment Spray pump will be used to restore SDC is also not correct. This is plausible because the Loss of Shutdown Cooling procedure directs stopping CORE ALTERATIONS. The examinee may not understand that the refueling procedure and Technical specifications state that suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. And use of a Containment Spray pump to restore SDC is an option in the Loss of Shutdown Cooling procedure.
C. Correct.
D. Incorrect. The "A" Containment Spray pump will be used to restore SDC is also not correct. This is plausible because the use of a Containment Spray pump to restore SDC is an option in the Loss of Shutdown Cooling procedure.

Technical Reference(s):

1. AOP 2572 Loss of Shutdown Cooling, rev. 020, pages 3 and 7.
2. BKG AOP 2572 Loss of Shutdown Cooling Background Document, rev. 020, page 17.
3. OP 2209A Refueling Operations, rev. 035, page 6, definitions.

Provided reference(s): None
Learning Objective: 283360 ILT Describe the conditions, and the basis for those conditions, necessary to start a LPSI pump as specified in AOP 2572, Loss of Shutdown Cooling.
Source: New
History: None
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments: NRC feedback 7-28. Feedback was to provide a new K/A that was applicable to the ROs. ROs do not perform refueling activities. New K/A A2.07 was provided on 7-29-22. New K/A A2.06 was provided on 8-9-22 due to K/A A2.07 overlap with Q91.

Question #61

The plant is operating at 80% power.

- Instrument Air is lost to CN-218, Condensate Minimum Flow Recirculation Control Valve.

How will the plant respond?

A. The plant will trip due to Low S/G Level.

B. S/G levels will fluctuate and stabilize at a lower level.

C. S/G levels will fluctuate and stabilize at a higher level.

D. There will be no appreciable effect on the plant.

Question #61 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A # 056 K1.03 Importance Rating: 3.8

K/A Statement: Knowledge of the physical connections and/or cause and effect relationships between the Condensate System and the following systems: MFW

Proposed Answer: **D**

Justification: Valve 2-CN-218 is normally aligned to maintain ~3200 gpm condensate minimum flow. This ensures condensate minimum flow after a plant trip. The valve fails closed on a loss of Instrument Air. Since there is 18,000 gpm condensate flow at 100% power, the valve is expected to be closed during power operations. A loss of air would therefore have no appreciable effect on either condensate or main feed.

Plausibility:

A. The student may think that the valve fails open, causing the S/G levels to lower to the trip setpoint. The minimum flow valve is not large enough to rob too much flow from the Main Feed system.
B. The student may think that the valve fails open, then flow robbed from the Main Feed system will cause S/G levels to lower. The Feedwater control system will maintain levels at setpoint.
C. The student may think when the valve fails closed, the increased feed flow will cause S/G levels will rise. The Feedwater control system will maintain levels at setpoint.
D. Correct

Technical Reference(s): Condensate Syatem Lesson Plan CON-01-C R10c2

Provided reference(s): None

Learning Objective:281044 – Describe how a transient or malfunctioning each of the following could affect operation of the Condensate system: D) Minimum recirculation flow

Source: Modified Bank #412211

History:

Cognitive Level: Comprehension or Analysis

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was to provide a new K/A since a discriminating question could not be written. Kept the question as is but 'beefed up' the justification. DF 8/12/22
Removed note that CN-218 fails closed based on keeping all the answers plausible. DF 8/22/22

Question #62

Annunciator CH “C” CTMT AIR GASEOUS RADIATION HI (C-01, C-28), is alarming.

ESAS responds by sending a CLOSE signal to what valves?

A. The outboard CTMT Purge Valves only

B. The inboard CTMT Purge Valves only

C. All CTMT Purge Valves

D. No CTMT Purge Valves

Question #62 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #072 K5.03 Importance Rating: 3.6

K/A Statement: Knowledge of the operational implications or cause and effect relationships of the following concepts as they apply to the Area Radiation Monitoring System: Containment isolation

Proposed Answer: **C**

Justification: If any of the 4 channels of CTMT Atmospheric Radiation Monitors (CAMS) alarms, ESAS will close ALL the CTMT Purge valves (CPVIS is a 1/4 logic). We consider this question to meet the K/A since the CAMS system is a de facto Area Rad Monitor as they monitor a specific area.

Plausibility:

A. The student may think the Channel 1 CAMs closes outboard isolations.
B. The student may think the Channel 1 CAMs closes inboard isolations.
C. Correct answer
D. The student may think the CPVIS is a 2/4 logic

Technical Reference(s): ARP 2590A-111, CH "C" CTMT AIR GASEOUS RADIATION HI

Provided reference(s): None

Learning Objective:

Source: New

History: N/A

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(5): Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments: NRC feedback 7-28. Feedback was to change stem of the question to "ESAS responds by sending a CLOSE signal to what valves". Change completed on 8-4-22. Changed

Question #63

The following conditions are noted at FLP-6, West DC Switchgear Room Fire Panel:

- Strobe lights and horn are pulsating quickly.

What is the status of the West DC Switchgear Room Fire system?

A. Only ONE smoke detector (photoelectric OR ion) has activated.

B. TWO smoke detectors (photoelectric AND ion) have activated.

C. The Halon system is discharging into the room.

D. The Halon system has completed discharging into the room.

Question #63 ☒ RO ☐ SRO

Tier # 2 Group # 2 K/A #086 A1.06 Importance Rating: 3.0

K/A Statement: Ability to predict and/or monitor changes in parameters associated with operation of the Fire Protection System, including: Lights and alarms

Proposed Answer: **B**

Justification: Activation of another smoke detector of opposite type (ion or photoelectric), causes the following to occur: Locally strobe and horns pulses quickly

Plausibility:

A. This condition causes a slow pulse of strobes and horn; student make think otherwise.
B. Correct
C. This condition causes a steady strobe and horn. The student may think otherwise
D. There is no change of condition for the Halon depleting.

Technical Reference(s): ARP 2590I, Alarm Response for Fire Panel, C---26, ZONE 45

Provided reference(s): None

Learning Objective: 287608 - State the purpose and describe the operating characteristics of the following major Fire Protection system components: G) Local Fire Detection Panels

Source: ☐ Modified Bank #451537

History:2001 NRC exam, Q78

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(7): Design, components, and functions of safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments: NRC feedback 7-28. Feedback was that answer “A” was a subset of answer “B”, so make answer “A” “Only ONE…”. Change completed on 8-4-22.

Question #64

What are the **MINIMUM** requirements for the Operator At The Controls (OATC) for conducting a control board walk down, in accordance with OP-AA-100, Conduct of Operations?

A. Walk down the entire control board a minimum of once per hour.

B. Walk down C01 through C04 control boards a minimum of once per hour.

C. Walk down the entire control boards, as needed, for proper monitoring.

D. Walk down the entire control boards prior to taking the watch.

Tier # 3 Category # 1 K/A #G2.1.1 Importance Rating: 3.8
K/A Statement: Knowledge of the conduct of operations requirements

Proposed Answer: **A**
Justification: Walk down the entire control boards a minimum of once per hour is correct. OP-AA-100, Conduct of Operations specifies that the OATC walks down the entire control board a minimum of once per hour.
Plausibility:

A. Correct.
B. Incorrect: Walk down C01 through C04 control boards a minimum every hour is not correct. This is plausible because C01 through C04 primary control boards are generally the responsibility of the OATC for operating and monitoring. The BOP generally monitors and operates C05 through C08 secondary control boards.
C. Incorrect: Walk down the entire control boards, as needed, for proper monitoring is not correct. This is plausible because this would be reasonable.
D. Incorrect: Walk down the entire control boards prior to taking the watch is not correct. This is plausible because the examinee might think the requirement to maintain continuous awareness of plant parameters covers control board walk downs and there is no requirement for control board walk downs other than prior to the watch.

Technical Reference(s): OP-AA-100, Conduct of Operations, rev. 45, page 26.
Provided reference(s): None
Learning Objective: 314290 Describe the responsibilities of the "Operator at the Controls".
Source: Bank #455541
History:
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments: NRC feedback 7-28. Feedback was to provide a new K/A since the original K/A was not a match for a tier 3 generic. Replaced question with a bank question. Change completed 8-8-22. NRC feedback on 8-22-22 was to make answers active with verbs and remove no time specified in answer "C". Changed answer "B" to specify the Reactor Operator panels since this was vague. Change completed 8-23-22.

Question #65

In accordance with OP 2260, Unit 2 User's Guide.

WHICH statement is accurate for communicating CAUTIONS and NOTES during the performance of an Emergency Operating Procedure (EOP)?

A. The US shall read aloud any CAUTIONS and NOTES, then communicate if the CAUTION or NOTE is applicable. The ROs respond by stating "Understand the NOTE or CAUTION".

B. The US shall read aloud all applicable CAUTIONS and NOTES in their entirety the first time encountered. The ROs perform a verbatim repeat back of the NOTE or CAUTION.

C. The US shall read aloud all applicable CAUTIONS and NOTES in their entirety the first time encountered. The ROs will paraphrase back the NOTE or CAUTION.

D. The US shall paraphrase aloud all applicable CAUTIONS and NOTES.
The ROs perform a verbatim repeat back of the US's paraphrased statement.

Question #65 ☒ RO ☐ SRO

Tier # 3 Category # 1 K/A #G2.1.32 Importance Rating: 3.8
K/A Statement: Ability to explain and apply system precautions, limitations, notes, or cautions

Proposed Answer: **C**
Justification: The US shall read aloud all applicable CAUTIONS and NOTES in their entirety the first time encountered. The ROs will paraphrase back the NOTE or CAUTION. This is correct and has delineated in OP 2260, Unit 2 User's Guide, step 1.7.2. This K/A is met because the question involves generic application of notes and cautions in EOPs.
Plausibility:

A. Incorrect: Plausible because it is reasonable that the US would read aloud the CAUTIONS and NOTES and then state whether there are applicable and the ROs would respond by stating "understand the NOTE or CAUTION".
B. Incorrect: Plausible because it is reasonable verbatim repeat back would be required.
C. Correct.
D. Incorrect: Plausible because it is reasonable that the US would paraphrase aloud the applicable CAUTIONS and NOTES and then the ROs would repeat back verbatim the US's paraphrased statement.

Technical Reference(s): OP 2260, Unit 2 User's Guide, rev. 017, page 11.
Provided reference(s): None
Learning Objective: 283639 ILT Given a set of plant conditions, apply the EOP usage rules as described in OP 2260, Unit 2 EOP User's Guide.
Source: Bank 415064
History: No history of being on a NRC exam
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments: NRC feedback 7-28. Feedback was the question does not meet the requirements for a tier 3 generic because it was a system question. Should be a procedure use and adherence question. Replaced question. Completed change 8-10-22.

Question #66

A Technical Specification Quarterly (92 days) surveillance was last performed 95 days ago.

1. What is the status of the equipment?

   AND

2. The equipment will remain in its current status ….

A. 1. OPERABLE
   2. as long as the surveillance is completed SAT within an additional 25% of the 92 days.

B. 1. OPERABLE
   2. as long as the surveillance is completed SAT within the next 92 days.

C. 1. NOT OPERABLE
   2. until the missed surveillance is completed SAT and a risk evaluation is performed within 24 hours.

D. 1. NOT OPERABLE
   2. until the missed surveillance is completed SAT.

Question #66 ☒ RO ☐ SRO

Tier # 3 Category # 2 K/A #2.2.12 Importance Rating: 3.7
K/A Statement: Knowledge of surveillance procedures

Proposed Answer: A
Justification: Operable in accordance with LCO 4.0.2. A maximum extension, not to exceed 25%, of the surveillance time (25% of 92 days is 115 days) is allowed.
Plausibility:

A. Correct
B. Incorrect. The surveillance must be completed SAT within 25% to remain OPERABLE in this condition. This is plausible because the examinee may confuse the use of LCO 4.0.2 with 4.0.3.
C. Incorrect. The equipment is still OPERABLE. The surveillance must be performed but 24 hours is not the limit. LCO 4.0.2 allows a maximum allowable extension not to exceed 25% of the surveillance time interval; 25% of 92 days (quarterly) is 115 days. And a risk evaluation is not required. This is plausible because the examinee may understand LCO 4.0.3 and only remember the 24 hour requirement. LCO 4.0.3 states that 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 specified surveillance interval, whichever is greater.
D. Incorrect. The equipment is still OPERABLE. A maximum extension, not to exceed 25%, of the surveillance time (25% of 92 days is 115 days) is allowed. This is plausible because the examinee may reason that failing to meet the surveillance frequency requirement deems the equipment inoperable. This would be the case if the equipment failed the surveillance.

Technical Reference(s): Technical Specifications 4.0.2 and 4.0.3.
Provided reference(s): None
Learning Objective: 282396 –As given in Technical Specifications, state the maximum allowed extension of a surveillance interval and the principles governing the use of an extension.
Source: Bank 487323
History: NRC 2020 Question 68.
Cognitive Level: Comprehension or Analysis
10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments: NRC feedback 7-28. Feedback was that answer “B” was a subset of answer “A”. Added “SAT” to answers and revised answer “B”. Change completed on 8-4-22.

Question #67

Which practice fulfills the requirements for Tags Plus?

A. A clam shell is tye wrapped over the tagged valve handwheel.

B. The tagged component is located within a Contaminated Area.

C. A sign is hung from the tagged component reading “Do Not Touch”.

D. Perform a Live-Dead-Live test on the tagged component prior to performing work.

Question #67 ☒ RO ☐ SRO

Tier # 3 Category # 2 K/A #2.2.13 Importance Rating: 4.1
K/A Statement: Knowledge of tagging and clearance procedures

Proposed Answer: **A**
Justification: Tags Plus utilizes a physical device or equivalent measure to prevent a single manipulation from admitting hazardous energy into a work area. The tye wrapped clam shell prevents manipulating the tagged component thereby preventing hazardous energy from entering the work area.
Plausibility:

A. Correct.
B. Access and operation of the component is not prevented. The student may think restricted access to a room may prevent operation of the component.
C. The sign does not prevent operation of the component. The student may think an administrative procedure can prevent operation of a component.
D. The test does not prevent energizing the circuit. The student may think a live-dead-live test ensures the work area is de-energized (it does but a breaker or switch can be operated after the test).

Technical Reference(s): OP-AA-200, Equipment Clearance, Attachment 17
Provided reference(s): none
Learning Objective: 325794 - Describe the requirements for 'Tags Plus'. Tags Plus Guidelines
Source: New
History:
Cognitive Level: Fundamental/memory
10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility
Comments: NRC feedback 7-28. Feedback was to add to answer "A" that that the component is tagged; which will make this answer correct, and change "device" to "component" in answers "B" and "C". Also to change cognitive level from comprehension/Analysis to fundamental/memory. Change completed on 8-4-22.

Question #68

The plant is operating at 100% power.

- Alarm 'VENT STACK RADMONITOR HI/FAIL' (C-06/7, A-12) annunciates.
- 'STACK GAS EFFLUENT', RIC 8168 'EQUIP FAIL' light is lit and cannot be cleared.

1. RM-8132B, Unit 2 Stack Gaseous Radiation Monitor, readings are considered ___1.___.

   AND

2. The above Annunciator Response Procedures (ARPs) direct ___2.___ RM-8132 purge.

A. 1. reliable
   2. starting

B. 1. reliable
   2. securing

C. 1. NOT reliable
   2. starting

D. 1. NOT reliable
   2. securing

Question #68 ☒ RO ☐ SRO

Tier # 3 Category # 3 K/A #G2.3.12 Importance Rating: 3.2

K/A Statement: Knowledge of radiological safety principles and procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, or alignment of filters

Proposed Answer: D

Justification: The conditions indicate a failure of RM-8168 which will cause RM-8132 to be isolated and purged with air. In this condition, RM-8132 readings are not reliable. The ARP states if RM-8168 fails, the PURGE VALVE is placed in the DISABLE position (which will stop RM-8132 from being purged and allow the radiation level to be monitored) in order to maintain monitoring capability of the Unit 2 stack.

Plausibility:

A. This is plausible if the student doesn't remember that RM-8132 will shift to being purged when RM-8168 is in a FAIL or HI condition. And that to restore RM-8132 so that it is reading accurately that purge must be stopped by taking the Purge Valve switch to DIASABLE.
B. This is plausible if the student doesn't remember that RM-8132 will shift to being purged when RM-8168 is in a FAIL or HI condition.
C. This is plausible because the student may think that enabling purge is what is needed for RM-8132 to read accurately.
D. Correct

Technical Reference(s): ARP 2590E-061, VENT STACK RADMONITOR HI/FAIL

Provided reference(s): None

Learning Objective: 282100 – Describe the automatic actions and logic associated with the Unit 2 Stack radiation Monitoring Subsystem

Source: Modified Bank #477170

History: NRC-2001, NRC-2008

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(11): Purpose and operation of radiation monitoring systems, including alarms and survey equipment

Comments: NRC feedback 7-28. Feedback was to add to revise question stem to ...The Unit 2 Stack Radiation Gaseous Monitor, RM-8132B, IS or IS NOT reliable and The Annunciator Response Procedure (ARP) above directs purge be started or secured. And to change the first answer for the question to either reliable or not reliable.

Question #69

Millstone 2 has entered and is working through an Abnormal Operating Procedure (AOP).

The plant trips. EOP 2525, Standard Post Trip Actions has been completed and an Optimal Recovery Emergency Operating Procedure (EOP) has been entered.

OP 2260, Unit 2 EOP User's Guide directs procedure usage as follows:

A. The EOP is the controlling procedure. AOPs can be used as directed by the US.

B. The US will designate the AOP or EOP as the controlling procedure.

C. The EOP is the controlling procedure. AOPs can no longer be used.

D. Neither the EOP nor AOP are the controlling procedure. The EOP and AOP will be worked in parallel.

Question #69 ☒ RO ☐ SRO

Tier # 3 Category # 4 K/A #G2.4.16 Importance Rating: 3.5

K/A Statement: Knowledge of emergency and abnormal operating procedures implementing hierarchy and coordination with other support procedures or guidelines, such as operating procedures, abnormal operating procedures, or severe accident management guidelines.

Proposed Answer: **A**

Justification: The EOP is always the controlling procedure. AOPs can be used as directed by the US. This is correct. OP 2260, Unit 2 EOP User's Guide, step 1.9.4.b states "Other procedures, including Normal OPs, Alarm Response Procedures (ARPs), and AOPs are not normally needed to supplement the EOPs, but may be used as directed by the US. The EOP is always the controlling procedure when used in conjunction with other procedures". This question meets the K/A since it asks about implementing hierarchy (controlling procedure between EOPs and AOPs) and coordination (how AOPs are used with EOPs).

Plausibility:

A. Correct.
B. Incorrect: The US will determine the most immediate need and designate the AOP or EOP as the controlling procedure is not correct. This is plausible because the US will prioritize actions and it is reasonable that they could designate any procedure that is addressing the immediate need as the controlling procedure. Examinee may not know usage standard.
C. Incorrect: The EOP is always the controlling procedure is correct. No other procedures are required since EOPs cover any actions required is not correct. The is plausible because the EOP set has appendices that cover most actions that maybe required. Examinee may not know usage standard.
D. Incorrect: Neither the EOP or AOP are the controlling procedure is not correct. The EOP and AOP will be worked in parallel could be correct. This is plausible because the US will prioritize actions and it is reasonable that they could designate any procedure that is addressing the immediate need as the controlling procedure. Examinee may not know usage standard.

Technical Reference(s): OP 2260, Unit 2 EOP User's Guide, rev. 017, page 16, step 1.9.4.b.

Provided reference(s): None.

Learning Objective: 283640 ILT Outline the rules of usage for the parallel performance of EOP and AOP actions as described in OP 2260, Unit 2 EOP User's Guide.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.41(b)(10): Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: New K/A 2.4.16 was provided on 8-9-22. A generic question could not be written for the original K/A; 2.4.2 Knowledge of system setpoints, interlocks, and automatic actions associated with emergency and abnormal operating procedure entry conditions. Change completed. New question written 8-11-22.

Question #70

The plant is operating at 100% power.

- The #1 Atmospheric Dump Valve (ADV) fails open.

The ADV opening causes the number of neutrons present in the core to be ___1.___ the number of neutrons in the previous generation AND $K_{eff}$ is ___2.___ .

A. 1. less than
   2. less than one (1)

B. 1. less than
   2. greater than one (1)

C. 1. greater than
   2. less than one (1)

D. 1. greater than
   2. greater than one (1)

Question #70 ☒ RO ☐ SRO

Tier # 4 Category: Reactor Theory K/A #192002 K1.07 Importance Rating: 3.1
K/A Statement: Define $K_{eff}$ and discuss its relationship to the state of a reactor (critical, subcritical, and supercritical)

Proposed Answer: **D**
Justification: $K_{eff}$ is defined as the ratio of the neutrons in one generation to the number in the previous generation. With the reactor at 100% power, the number of neutrons from one generation to the next are the same. When the Atmospheric Dump Valve (ADV) fails open, RCS temperature lowers and positive reactivity is added to the reactor as a result of the decreased temperature of the moderator. The addition of positive reactivity to the core results in increased fission which produces more neutrons. The increase in neutrons results in $K_{eff}$ being greater than one, since there are now more neutrons than the previous generation.
Plausibility:

A. The examinee may not understand the affect the ADV opening has on the reactivity of the core. This condition indicates a sub-critical reactor.
B. The examinee may not understand the affect the ADV opening has on the reactivity of the core and may not understand the relationship between the reactivity of the core and $K_{eff}$.
C. This condition indicates a supercritical reactor. The examinee may not understand the relationship between the reactivity of the core and $K_{eff}$.
D. Correct.

Technical Reference(s):
Provided reference(s): none
Learning Objective: GFES – Define the following terms associated with the neutron life cycle:
a Effective Multiplication Factor

Source: New
History:
Cognitive Level: Comprehension or Analysis
10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects
Comments: NRC feedback 7-28. Feedback was to add “one (1)” to the second part of each answer and to remove “one (1)” from the question stem. Also to correct and add additional information to the justification. In addition, while not feedback, additional detail was added to the plausibility’s. Change completed on 8-4-22.

Question #71

Millstone Unit 2 is at 100% power and has been operating at this power level for 100 days.

1. State the primary production mechanism of xenon-135.

   AND

2. State the secondary production mechanism of xenon-135.

A. 1. Directly from fission
   2. Iodine decay

B. 1. Directly from fission
   2. Promethium decay

C. 1. Iodine decay
   2. Directly from fission

D. 1. Promethium decay
   2. Directly from fission

Question #71 ☒ RO ☐ SRO

Tier # 4 Category: Reactor Theory K/A #192006 K1.03 Importance Rating: 2.8
K/A Statement: Describe the production of xenon-135

Proposed Answer: **C**
Justification: Primarily from iodine decay, secondarily from fission. Xe -135 has two methods of production: about 95% of xenon is produced by the decay of iodine -135 (a fission product, the concentration of which is proportional to power), and the remaining 5% of the xenon results directly from fission.
Plausibility:

A. Student could think that most xenon is from fission and that iodine decay is a component of the production.
B. Student could think that most xenon is from fission and that promethium decay is a component of the production.
C. Correct Answer.
D. Student could think that most xenon is from promethium decay and secondarily from fission. And secondarily from fission is correct

Technical Reference(s): PWR Generic Fundamentals – Reactor Theory – Chapter 6 – Fission Product Poisons.
Provided reference(s): None
Learning Objective: Objective 3 of 192006; Describe the production of xenon-135.
Source: Bank #192006 QID 1559
History: This question was written from GFES Bank question P1559 (Millstone question 391122). The stem was modified to adapt to Millstone Unit 2.
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects
Comments: NRC feedback 7-28. Feedback was to add “directly from” to the fission answers. Change completed on 8-4-22.

Question #72

As reactor power is ___1.___, CEA insertion limits (PDIL, PPDIL) ensure that CEAs will be positioned higher in the core to compensate for the change in ___2.___.

A. 1. raised
   2. Xe effects of reactivity

B. 1. raised
   2. power defect

C. 1. lowered
   2. Xe effects of reactivity

D. 1. lowered
   2. power defect

Question #72 ☒ RO ☐ SRO

Tier # 4 Category: Reactor Theory K/A #192005 K1.17 Importance Rating: 3.9
K/A Statement: Discuss rod insertion limits

Proposed Answer: **B**
Justification: As power is raised, CEA insertion limits are raised to ensure shutdown margin is met due to the change in power defect (affected by fuel temperature and moderator temperature).
Plausibility:

A. The examinee might think that pulling CEAs to account for Xe building in has something to do with PDIL and PPDIL
B. Correct.
C. The examinee might think that the negative reactivity effects of Xe building in could lower PDIL and PPDIL.
D. The examinee might think that the negative reactivity effects of raising temperature (fuel and moderator) could lower PDIL and PPDIL.

Technical Reference(s): None
Provided reference(s): none
Learning Objective:
Source: Bank # 391217
History:
Cognitive Level: Comprehension or Analysis
10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects
Comments: NRC feedback 7-28. Feedback was to rewrite the stem to make the question clearer and the NRC provided the stem change. Change completed on 8-4-22.

Question #73

The plant is operating steady-state at 80% power with all CEAs fully withdrawn.

Compare a 20 step group CEA insertion to a 20 step individual CEA insertion.

Assume the following:

- Reactor power returns to 80%.
- The individual CEA is NOT the center CEA.

Inserting a group of CEAs will cause a ___1.___ effect on the Axial Shape Index and a ___2.___ effect on the Radial Peaking Factor when compared to inserting only an individual CEA.

A. 1. smaller
   2. smaller

B. 1. smaller
   2. larger

C. 1. larger
   2. smaller

D. 1. larger
   2. larger

Question #73 ☒ RO ☐ SRO

Tier # 4 Category: Thermodynamics K/A #190009 K1.07 Importance Rating: 3.3
K/A Statement: Describe factors that affect peaking and hot channel factors

Proposed Answer: C
Justification: Inserting a bank of rods will cause a larger effect of the axial peaking factor due to 'pushing' more power production towards the bottom of the core. The effect on radial peaking is smaller since the Group CEAs are equally positioned on the radial plane.
Plausibility:

A. The examinee could not understand the difference between radial peaking and axial peaking. This is a fairly difficult concept.
B. The examinee could not understand the difference between radial peaking and axial peaking. This is a fairly difficult concept.
C. Correct.
D. The examinee could not understand the difference between radial peaking and axial peaking. This is a fairly difficult concept.

Technical Reference(s): None
Provided reference(s): None
Learning Objective: (GFES) Describe power peaking and hot channel factors
Source: Bank #390457
History:
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(1): Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects
Comments: NRC feedback 7-28. Feedback was to rewrite the stem to make the question clearer and the NRC provided the stem change. Also, while not provided as feedback, the justification and plausibility statements were enhanced. Change completed on 8-4-22.

Question #74

The unit is shutdown with 2-AR-17, Condenser Vacuum Breaker, OPEN.

What is the pressure in the Condenser?

A. 0.0 mm Hg absolute

B. 14.7 psig

C. 14.7 psia

D. 30.0" $H_2O$ absolute

Question #74 ☒ RO ☐ SRO

Tier # 4 Category: Thermodynamics K/A #193001 K1.01 Importance Rating: 2.7
K/A Statement: Convert between absolute and gauge pressure and vacuum scales

Proposed Answer: **C**
Justification: With the vacuum breaker open, the condenser would be at atmospheric pressure (14.7 psia). And 14.7 psia = 0 psig.
Plausibility:

A. Incorrect: 0.0 mm $H_g$ is 0 psia. This is plausible because the examinee may think 0.0 mm Hg absolute is the same as atmospheric pressure
B. Incorrect: 14.7 psig is 29.4 psia. This is plausible because the examinee may confuse psig with psia.
C. Correct:
D. Incorrect: 30 inches of water is 1.083 psia. This is plausible because the examinee may confuse "$H_2O$ with "Hg and 30 inches of mercury is 14.7 psia.

Technical Reference(s): none
Provided reference(s): none
Learning Objective:
Source: New
History:
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.41(b)(14): Principles of heat transfer, thermodynamics, and fluid dynamics.
Comments: NRC comment (editorial) for pre-submittal 10 questions. NRC comment was to change H2O in answer D to $H_20$ and put into justification what 30" of $H_20$ correspondence to atmospheric pressure. Note that 30 inches of Hg = 14.7 psia and 30 inches of $H_2O$ = 1.083 psia. Changes incorporated 6/9 by jwr. Accepted by the NRC 7-28-22.

Question #75

The plant tripped from 100% power when offsite power was lost. Natural Circulation flow has been verified by the crew.

What will enhance Natural Circulation flow in the Reactor Coolant System?

A. Raising Pressurizer level.

B. Lowering Steam Generator Level.

C. Raising Pressurizer pressure.

D. Lowering Steam Generator pressure.

Question #75 ☒ RO ☐ SRO

Tier # 4 Category: Thermodynamics K/A #193008 K1.23 Importance Rating: 4.1
K/A Statement: Describe means by which natural circulation can be enhanced

Proposed Answer: **D**
Justification: Mass flow rate is proportional to ΔT. Lowering Steam Generator pressure will lower $T_{COLD}$ raise the ΔT across the core. The increased ΔT provides a great driving force to create flow in the RCS (colder, denser water forces the warmer, lighter water up through the core region).
Plausibility:

A. Raising PZR level has no effect on the flow through the RCS, the examinee may think raising level may raise pressure in the RCS, causing better flow through the system. In addition, the examinee could remember that the Pressurizer is required for Natural Circulation, as described in the Pressurizer TS bases. And, that to be Operable it must be within a certain level and must have heaters available for pressure control.
B. Lowering S/G level will not enhance RCS flow.
C. Raising PZR pressure has no effect on the flow through the RCS, the examinee may think raising pressure in the RCS, will cause better flow through the system. In addition, the examinee could remember that the Pressurizer is required for Natural Circulation, as described in the Pressurizer TS bases. And, that to be Operable it must be within a certain level and must have heaters available for pressure control.
D. Correct.

Technical Reference(s): GFES PWR Thermal Hydraulics
Provided reference(s): None
Learning Objective: GFES Describe the means by which the operator can enhance natural circulation

Source: New
History:
Cognitive Level: Comprehension or Analysis
10CFR55.41(b)(14): Principles of heat transfer, thermodynamics, and fluid dynamics.
Comments: NRC feedback 7-28. Feedback was to edit the stem to say …flow has been verified… and enhance the plausibility statement for answer "A". Also, while not provided as feedback, the plausibility statement for answer "C" was enhanced. Change completed on 8-4-22.

Question #76

The plant tripped from 25% power.

1. What chemistry sample is required by Technical Specifications?

   AND

2. The basis for this sample is to minimize dose consequences for what event?

A. 1. RCS Specific Activity
   2. SGTR

B. 1. Containment Atmosphere grab sample
   2. SGTR

C. 1. RCS Specific Activity
   2. LOCA

D. 1. Containment Atmosphere grab sample
   2. LOCA

Question #76 ☐ RO ☒ SRO

Tier # 1 Group # 1 K/A #007 G 2.1.34 Importance Rating: 3.5

K/A Statement: Reactor Trip. Knowledge of RCS or balance-of-plant chemistry controls, including parameters measured and reasons for the control

Proposed Answer: **A**

Justification: TS 3.4.8, SPECIFIC ACTIVITY, requires a Iodine Specific Activity sample to be taken within 2 to 6 hours after a power change of greater than 15% in one hour. The sample is based on Iodine spikes combined with a MSLB or SGTR affecting off-site and CRE dose rates

Plausibility:

A. Correct Answer
B. CTMT grab samples are required in the event of a failure of CTMT atmospheric radiation monitors (TS3.4.6.1, LEAKAGE DETECTION SYSTEMS)
C. RCS Specific Activity limits are based on MSLB or SGTR. Exceeding Tech Spec. High Activity is not a concern during a LOCA since the activity is confined to the CTMT building.
D. CTMT grab samples are required in the event of a failure of CTMT atmospheric radiation monitors (TS3.4.6.1, LEAKAGE DETECTION SYSTEMS)

Technical Reference(s): Millstone 2 T.S. 3.4.8, SPECIFIC ACTIVITY,
EOP 2541, Appendix 4B, Follow-up Actions

Provided reference(s): None

Learning Objective: 283656 – Outline the subsequent actions for EOP 2525, Standard Post-Trip Actions

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases

Comments: Accepted by the NRC 7-28-22.

**Question #77 was administered as part of the 100-question exam. The licensee's post-exam comments identified that the question had no correct answer. NRC concurred with the licensee. The question was deleted from the exam.**

~~Question #77~~

~~A Small Break Loss of Coolant Accident has occurred.~~

~~EOP 2532, Loss of Coolant Accident procedure requires a controlled cooldown be performed.~~

~~1. The cooldown is required to be initiated __1.__ .~~

~~AND~~

~~2. The reason for cooldown is to __2.__ ?~~

~~A. 1. after 1 hour at > 40 °F/hr and the RCS be reduced by at least 40 °F after 2 hours~~
~~2. conserve Condensate inventory and comply with the long term cooling analysis~~

~~B. 1. after 1 hour at > 40 °F/hr and the RCS be reduced by at least 40 °F after 2 hours~~
~~2. to ensure Safety Injection maintains adequate Shutdown Margin~~

~~C. 1. within 1 hour at a rate of > 40 °F/hr~~
~~2. conserve Condensate inventory and comply with the long term cooling analysis~~

~~D. 1. within 1 hour at a rate of > 40 °F/hr~~
~~2. to ensure Safety Injection maintains adequate Shutdown Margin~~

Question #77 ☐ RO ☒ SRO

Tier # 1 Group # 1 K/A #008 G2.4.18 Importance Rating: 4.0

K/A Statement: Pressurizer Vapor Space Accident. Knowledge of the specific bases for emergency and abnormal operating procedures

Proposed Answer: A

Justification: The cooldown is required to be initiated after 1 hour at > 40 °F/hr and the RCS be reduced by at least 40 °F after 2 hours. This is required per the note prior to step 19 of EOP 2532, Loss of Coolant Accident. The reason for the cooldown requirements is to conserve Condensate inventory and comply with the long term cooling analysis. Condensate inventory needs to be conserved if the condenser is unavailable since this will require the use of Auxiliary Feedwater and the Atmospheric Dump Valves.

Plausibility:

A. Correct.

B. Incorrect: The reason for cooldown is to ensure Safety Injection maintains adequate Shutdown Margin is not correct. This is plausible because cooling down would lower RCS pressure and raise Safety Injection flow and add borated water to the RCS.

C. Incorrect: The cooldown is required to be initiated within 1 hour at a rate > 40 °F/hr is not correct. This is plausible because operators have misinterpreted this cooldown note in the past as requiring cooldown to be established at > 40 °F/hr within 1 hour.

D. Incorrect: The cooldown is required to be initiated within 1 hour at a rate > 40 °F/hr is not correct. And the reason for cooldown is to ensure Safety Injection maintains adequate Shutdown Margin is not correct. The cooldown requirement is plausible because operators have misinterpreted this cooldown note in the past. The reason for the cooldown is plausible because cooling down would lower RCS pressure and raise Safety Injection flow and add borated water to the RCS.

Technical Reference(s):

1. EOP 2532, Loss of Coolant Accident, rev. 037, page 29.

2. EOP 2532 LOCA, Lesson Text Power Point, E32-01-C, rev. 5/1, pages 1, 41, and 42.

Provided reference(s): none

Learning Objective: 283789 ILT Outline and explain the bases for the major actions in EOP 2532, Loss of Coolant Accident.

Source: New

History:

Cognitive Level: Memory/Fundamental

10CFR55.43(b)(1): Conditions and limitations in the facility license.

~~Comments: NRC feedback 7-28. Feedback was to provide a new K/A since a discriminating question could not be written since there are really no actions that are performed outside the control room that are purely generic. Wrote a new question with the new K/A G2.4.18. Change completed 8-16-22.~~

Question #78

The plant experienced a SBLOCA. The following conditions currently exist:

- $T_{HOT}$ = 520 °F and stable.
- $T_{COLD}$ = 518 °F and stable.
- Pressurizer Pressure = 1325 psia and rising.
- Pressurizer Level = 33% and rising.
- RVLMS = 80% and stable.
- S/G levels = 30% and rising, feeding with AFW.

In accordance with EOP 2532, Loss of Coolant Accident…

1. Can the HPSI flow be throttled/stopped?

   AND

2. Why or why NOT?

A. 1. No
   2. RVLMS level of 80% is too low

B. 1. No
   2. S/G levels of 30% are too low

C. 1. Yes
   2. Pressurizer level, as well as, the other criteria are met

D. 1. Yes
   2. Loop ΔT, as well as, the other criteria are met

Question #78 ☐ RO ☒ SRO

Tier # 1 Group # 1 K/A #009 EA2.34 Importance Rating: 3.7

K/A Statement: Ability to determine and/or interpret the following as they apply to a Small-Break LOCA: Conditions for throttling or stopping HPI

Proposed Answer: **C**

Justification: Yes, HPSI flow can be stopped because Pressurizer level, as well as other criteria are met. EOP 2532, LOCA, step 24 specifies HPSI Throttle criteria as: 1) subcooling above the minimum operating limit of the RCS P/T curve based on CETs OR $T_H$ (> 30 °F), 2) Pressurizer level is greater than 20% AND stable OR rising, 3) At least one S/G is available for heat removal AND S/G level meets ONE of the following , 40-70% or level being restored by main feedwater, Aux feedwater, or Condensate pump, and 4) Reactor vessel level is greater than or equal to 43%. All these conditions are met. K/A is matched because the examinee is required to know the requirements for throttling HPSI. SRO knowledge of when to perform steps and bases of how steps are performed from guidance documents.

Plausibility:

A. Incorrect. HPSI throttle criteria is met. RVLMS meets the HPSI throttle criteria of greater than or equal to 80%. This is plausible because the examinee may not know the HPSI throttle criteria and may reason that the reactor needs to be full before HPSI (Safety Injection) can be stopped.
B. Incorrect. HPSI throttle criteria is met. This is plausible because the examinee may not know the HPSI throttle criteria or may know that one of the things that satisfy the S/G level criteria is level 40 – 70% but not know the other criteria of level being restored by main feedwater, Aux feedwater, or Condensate pump.
C. Correct.
D. Incorrect. HPSI throttle criteria is met but there is no Loop ΔT requirement as part of HPSI throttle criteria. This is plausible because Loop ΔT is criteria for Natural Circulation flow and the examinee could confuse it with HPSI throttle criteria.

Technical Reference(s):

1. EOP 2532, Loss of Coolant Accident, Revision 037, pages 33 and 43.
2. OP 2260, EOP Users Guide, Rev. 017, page 45.

Provided reference(s): None

Learning Objective: 283792 – ILT Given a set of plant conditions concerning a Loss of Primary Coolant determine if criteria for the following are met: B. HPSI/LPSI Termination.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: NRC feedback 7-28. Feedback was to rewrite the question; remove first bullet for CEAs, rewrite question stem to ask can HPSI be throttled…yes or no… and why or why not, and change second part of answers to match new question. Also changed justification and plausibility statements to match answers. Change completed on 8-5-22.

Question #79

The plant is in Reduced Inventory Operations with the following plant conditions:

- Pressurizer vent port is removed.
- L-112, RCS WIDE RANGE LEVEL = 0".
- L-122, RCS NARROW RANGE LEVEL = 0".
- Shutdown Cooling Flow = 1400 gpm.

A PEO reports "abnormal knocking and banging noises" emanating from the 'A' LPSI pump.

- 'A' LPSI pump current = 15 amps and oscillating.

1. What procedure is used to mitigate this issue?

   AND

2. What actions are taken?

A. 1. OP 2310, Shutdown Cooling
   2. Throttle the LPSI injection valves.

B. 1. OP 2310, Shutdown Cooling
   2. SECURE the 'A' LPSI pump.

C. 1. AOP 2572, Loss of Shutdown Cooling
   2. Throttle the LPSI injection valves.

D. 1. AOP 2572, Loss of Shutdown Cooling
   2. SECURE the 'A' LPSI pump.

Question #79 ☐ RO ☒ SRO

Tier # 1 Group # 1 K/A #025 AA2.07 Importance Rating: 4.0

K/A Statement: Ability to determine and/or interpret the following as they apply to the Loss of the Residual Heat Removal System: Pump cavitation

Proposed Answer: **D**

Justification: The examinee should recognize that the 'A' LPSI pump is cavitating by the local noises and low (and oscillating) amps. AOP 2572 considers these as indicators to an imminent loss of SDC. The procedure (Step 12) directs the running pump to be tripped if the LPSI pump is cavitating.

Plausibility:

A. The examinee may think that the SDC procedure may contain steps to address the 'A' LPSI pump issue. Since the pump is still operating and flow is good, they may think reducing system flow will mitigate the issue.
B. The examinee may think that the SDC procedure may contain steps to address the 'A' LPSI pump issue.
C. AOP 2572 allows starting the second LPSI pump under certain conditions. Since the pump is still operating and flow is good, they may think reducing system flow will mitigate the issue.
D. Correct.

Technical Reference(s): AOP 2572, Loss of SDC
OP 2310, Shutdown Cooling

Provided reference(s): None

Learning Objective: Evaluate plant conditions to determine AOP 2572, Loss of Shutdown Cooling, applicability

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: 18MAY22 changed RCS flowrate to 1400 gpm, the minimum flowrate allowed for RIO. NRC feedback 7-28. Feedback was to change the stem question to something different than …describe your actions? Also, while not a feedback item, answers "A" and "B" were changed from "Enter" to "Stay in", since the crew would be in this procedure when operating SDC. Change completed on 8-5-22. Based on validator (A Leone) feedback, modified the question and answers to clarify what procedure has the actions to mitigate the issue. DF 8/12/22.

Question #80

The plant is operating at 100% power.

Numerous alarms annunciate coincident with a loss of the following indications:

- LPSI FLOW TO LOOP 2A (FT-332)
- LPSI FLOW TO LOOP 2B (FT-342)
- The Core mimic
- The 'B' SGFP insert

1. What procedure will be entered to address the loss of power?

   AND

2. What actions will be taken to mitigate the condition?

A. 1. AOP 2504B, Loss of Non-Vital Instrument Bus VR-21
   2. Take Local Manual control of FW-51B, #2 S/G Feedwater Regulating valve.

B. 1. AOP 2504B, Loss of Non-Vital Instrument Bus VR-21
   2. Close Trip Circuit Breakers (TCBs) #3, #4, #7, and #8.

C. 1. AOP 2504F, Loss of Vital Instrument Bus VA-40
   2. Take Local Manual control of FW-51B, #2 S/G Feedwater Regulating valve.

D. 1. AOP 2504F, Loss of Vital Instrument Bus VA-40
   2. Close Trip Circuit Breakers (TCBs) #3, #4, #7, and #8.

Question #80 ☐ RO ☒ SRO

Tier # 1 Group # 1 K/A #057 G2.2.44 Importance Rating: 4.4

K/A Statement: Loss of Vital AC Electrical Instrument Bus. Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions

Proposed Answer: A

Justification: The numerous alarms and loss of indications indicate a loss of an electrical bus. In this case, the lost bus is Non-Vital Instrument bus VR-21 which powers the right side of C-01 and the core mimic (AOP 2501 diagnostic note). A loss of VR-21 will cause the #2 S/G FRV to 'lock up', requiring an operator to be dispatched to take local manual control of the valve to maintain S/G levels (and close the valve in case of a plant trip).

Plausibility:

A. Correct.
B. Incorrect. Closing TCBs is not correct. The TCBs will open on a loss of Vital Instrument Bus VA-40. The examinee may think since the TCBs are on the right side of the TCB cluster, VR-21 may power the buttons.
C. Incorrect. The examinee may think VA-40 is lost and AOP 2504F, Loss of Vital Instrument Bus VA-40 would be appropriate to address the casualty. The B SGFP insert is powered (auctioneered) from both VA-40 and VR-21 but only VR-21 powers the lights.
D. Incorrect. The examinee may think VA-40 is lost and AOP 2504F, Loss of Vital Instrument Bus VA-40 would be appropriate to address the casualty. The B SGFP insert is powered (auctioneered) from both VA-40 and VR-21 but only VR-21 powers the lights. If VA-40 were lost, the TCBs #3, #4, #7, and #8 would open (although the plant would not trip)

Technical Reference(s):

1. AOP 2501, Diagnostic for Loss of Electrical Power, Attachment 1, Lost Control Power.
2. AOP 2504B, Loss of Non-Vital Instrument Bus VR-21

Provided reference(s): None

Learning Objective: 280945 – Define the following Annunciator System terms and conditions in accordance with OP 2387A: (a) Annunciator Out of Service.

Source: Modified Vision # 487544

History: 2020 NRC exam #91

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: NRC feedback 7-28. Feedback was to provide a new K/A since the original K/A was not a match for an AOP abnormal and therefore a discriminating question that matched the K/A could not

be written. Modified question #487544 from a loss of VR-11 (compared to a loss of VA-30) to a loss of VR-21 (compared to a loss of VA-40). Changed distractors to reflect opening of 4 TCBs on a loss of VA panel. Added core mimic to list of indications, removed HPSI flow from list of indicators. DF 8/12/22.

Question #81

The plant is operating at 100% power.

A major air leak occurs.

1. What Instrument Air system automatic actions occur as air pressure lowers?

   AND

2. What LCO would be affected by a complete loss of Instrument Air?

A. 1. Station Air Supply to Instrument Air (2-SA-10.1) opens.
   2. 3.7.1.7, Atmospheric Dump Valves

B. 1. Station Air Supply to Instrument Air (2-SA-10.1) opens.
   2. 3.7.1.6, Main Feedwater Isolation Components

C. 1. Instrument Air Supply to Station Air (2-IA-641) isolates.
   2. 3.7.1.7, Atmospheric Dump Valves

D. 1. Instrument Air Supply to Station Air (2-IA-641) isolates.
   2. 3.7.1.6, Main Feedwater Isolation Components

Question #81 ☐ RO ☒ SRO

Tier # 1 Group # 1 K/A #065 AA2.09 Importance Rating: 3.3

K/A Statement: Ability to determine and/or interpret the following as they apply to Loss of Instrument Air: Automatic IAS response as air header pressure lowers

Proposed Answer: **B**

Justification: During an air leak in the Instrument Air system, lowering pressure to 85 psig will cause the Station Air supply to Instrument Air, SA-10.1, to open and the Station Air supply to Station Air header, SA-11.1, to close. When air pressure gets too low (~60 psi), the Main Feedwater Regulating valves will lock up and won't have the motive force to close on a MSI signal, causing LCO 3.7.1.6., Main Feedwater Isolation Components to be NOT met.

Plausibility:

A. The Atmospheric Dump valves (ADVs) are still considered OPERABLE as long as they can be operated in local manual mode. The examinee may think the ADV need to be able to be operated from the control room to be OPERABLE.
B. Correct.
C. 2-IA-641, Instrument Air Supply to Station Air is a manually operated valve. The Atmospheric Dump valves (ADVs) are still considered OPERABLE as long as they can be operated in local manual mode. The examinee may think the ADV need to be able to be operated from the control room to be OPERABLE.
D. 2-IA-641, Instrument Air Supply to Station Air is a manually operated valve.

Technical Reference(s): AOP 2563, Loss of Instrument Air
MP2 Technical Specifications TS LCO 3.7.1.6 & 3.7.1.7

Provided reference(s): none

Learning Objective: Identify those IA/SA valves which automatically reposition due to lowering air pressure and explain the effects on the Instrument Air system.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(1): Conditions and limitations in the facility license.

Comments: This was one of the NRC pre-submittal 10 questions. NRC feedback 7-28. Feedback was to provide a new K/A since the original K/A was not a match for an AOP abnormal and therefore a discriminating question that matched the K/A could not be written. New question written. DF 8/15/22.

Question #82

Refueling is in progress.

- Neutron count rate is being monitored using channels “A”, “B”, and “C” wide range detectors.
- Channel “A” wide range detector is providing audible indication.

Vital instrument bus VA-10 is lost.

1. Can refueling continue without operator action?

   AND

2. What Technical Specification addresses the wide range neutron detectors in this condition?

A. 1. Yes
   2. TS 3.3.1.1, Reactor Protection Instrumentation

B. 1. Yes
   2. TS 3.9.2, Instrumentation

C. 1. No
   2. TS 3.3.1.1, Reactor Protection Instrumentation

D. 1. No
   2. TS 3.9.2, Instrumentation

Question #82 ☐ RO ☒ SRO

Tier # 1 Group # 2 K/A #032 AA2.01 Importance Rating: 3.0

K/A Statement: Ability to determine and/or interpret the following as they apply to Loss of Source Range Nuclear Instrumentation: Normal and/or abnormal power supply operation

Proposed Answer: D

Justification: Refueling can't continue and TS 3.9.2 is the applicable Technical Specification. TS 3.9.2 requires two source range neutron flux monitors be OPERABLE, each with continuous visual indication in the control room and one with audible indication in the containment, and control room. The requirement for one audible indication is not met with the loss of VA-10. Therefore, the action to immediately suspend all operations involving CORE ALTERATIONS must be taken. This TS is applicable in MODE 6 (refueling). This K/A is met in that it involves the ability to determine the Loss of Source Range Nuclear Instrumentation (a channel that being used for audible indication during refueling) with an abnormal power supply (loss of vital instrument bus VA-10). It meets the SRO only requirements in that the examinee must apply conditions (refueling, wide range channels, and channel selected for audible indications) and limitations (selection of applicable TS and action required).

Plausibility:

A. Incorrect: Refueling can't continue and TS 3.3.1.1 doesn't apply. Audible indication is loss when VA-10 is lost (de-energizes) and TS 3.9.2 requires fuel movement be immediately suspended. TS 3.3.1.1. is applicable in MODES 3, 4, and 5. This is plausible because the examinee may not know the Technical Specification for refueling and know that the Reactor Protection Instrumentation TS covers wide range instrumentation but may not know it's not applicable in MODE 6.
B. Incorrect: Refueling can't continue. Audible indication is loss when VA-10 is lost (de-energizes) and TS 3.9.2 requires fuel movement be immediately suspended. This is plausible because the examinee may not know that audible indication is required, but know that only two wide range detectors are required in MODE 6.
C. Incorrect: TS 3.3.1.1 doesn't apply in MODE 6. TS 3.3.1.1. is applicable in MODES 3, 4, and 5. This is plausible because the examinee may not know the Technical Specification for refueling and know that the Reactor Protection Instrumentation TS covers wide range instrumentation but may not know it's not applicable in MODE 6.
D. Correct.

Technical Reference(s): MP2 Technical Specifications TS LCO 3.9.2 and 3.3.1.1.

Provided reference(s): none

Learning Objective: 281829 ILT Given any operating condition for the Nuclear Instrumentation System, state whether the condition requires entry into Technical Specifications.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases.

Comments: NRC feedback 7-28. Feedback was that the question was not an SRO only question, since it was just knowledge. Make into an SRO question with tying it to a TSAS, below the line. Not sure exactly what feedback was. Wrote a new question addressing the K/A at the SRO only level. Completed question rewrite on 8-16-22.

Question #83

The plant is operating at 100% power.

- Fire Panel, C-26, ZONE 37, CONTAINMENT FIRE PANEL FLP-3, alarms
- FLP-3, ZONE 2, CTMT B RCP HEAT, is in alarm.
- FLP-3, ZONE 3, INSIDE CTMT EAST PENETRATION SMOKE, is in alarm.

The crew enters AOP 2559, Fire.

What components are isolated by AOP 2559, Fire?

A. #1 & #2 MSIVs, 2-MS-64A and 2-MS-64B

B. PORVs 2-RC-402 and 2-RC-404

C. #1 & #2 S/G BLDN ISOL VLVs 2-MS-220A and 2-MS-220B

D. Ch. 1 & 2 ADV QUICK OPEN PERMISSIVE signals

Question #83 ☐ RO ☒ SRO

Tier # 1 Group # 2 K/A #067 AA2.12 Importance Rating: 3.3

K/A Statement: Ability to determine and/or interpret the following as they apply to Plant Fire on Site: Location of vital equipment within fire zone

Proposed Answer: **B**

Justification: A fire in CTMT (Fire Area R-15) requires the PORVs to be isolated at the Bottle Up panels. SRO justification. Fires and maintaining equipment available in the event of a fire, including isolation of equipment (all answers listed), are conditions and limitations in the facility license. SROs through the use of AOP 2559, Fire and Appendix R procedures maintain equipment for safe shutdown.

Plausibility:

A. The MSIVs are closed during a fire in Fire Area R-1, the examinee may think the MSIVs need to be closed for a fire in CTMT
B. Correct answer
C. Blowdown is isolated on a fire in Fire Area R-2, the examinee may think Blowdown needs to be secured for a fire in CTMT
D. ADV Quick Open is disabled on a fire in Fire Area R-11, the examinee may think the Quick Open signal need to be isolated for a fire in CTMT

Technical Reference(s): AOP 2559, Fire
AOP 2579N, Fire Procedure for Hot Standby Appendix R Fire Area R-15.

Provided reference(s): None

Learning Objective: Outline the major actions for Shift Manger/Unit Supervisor using AOP 2559, Fire

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(1): Conditions and limitations in the facility license.

Comments: NRC feedback 7-28. Feedback was that this needs to be justified as a SRO. Discussed was that this was conditions and limitations in the facility license. That it is Appendix R. A SRO justification was added. Also removed “you” from the stem. Change completed 8-5-22.

Question #84

The plant is in MODE 3.

AOP 2568, Reactor Coolant System Leak is entered.

Plant conditions are as follows:

- Pressurizer pressure = 2250 psia and stable.
- Pressurizer level = 36% and stable.
- RCS temperature $T_{COLD}$ = 532 °F and stable.
- Two (2) Charging pumps are operating.
- Letdown flow = 28 gpm.

Pressurizer level and Pressurizer pressure begin slowly lowering. RCS temperature remains stable.

1. What do these plant conditions indicate?

   AND

2. What actions will the Unit Supervisor direct?

A. 1. The leak has become a LOCA.
   2. Perform AOP 2568 Attachment A, Manual SIAS Initiation While Shutdown.

B. 1. The leak has become a LOCA.
   2. Go to EOP 2532, Loss of Coolant Accident.

C. 1. A steam leak has developed.
   2. Refer to AOP 2589, Excessive Steam Load.

D. 1. A steam leak has developed.
   2. Go to EOP 2536, Excess Steam Demand Event.

Question #84 ☐ RO ☒ SRO

Tier # 1 Group # 2 K/A # A16 AA2.06 Importance Rating: 3.5

K/A Statement: Ability to determine and/or interpret the following as they apply to Excess RCS Leakage: RCS temperature and pressure

Proposed Answer: B

Justification: Pressurizer pressure lowering with temperature stable is an indication the leak has worsened. With two Charging pumps operating and letdown at 28 gpm (the low limit) the leak, the leak now exceeds the capacity of the Charging pumps. The leak is now considered a LOCA and EOP 2532, Loss of Coolant Accident with be entered. AOP 2568 Fold out page states if Pressurizer level is lowering with second charging pump running in MODE 3 AND SIAS is not blocked: GO TO EOP 2541 Appendix 1, Diagnostic Flowchart or EOP 2532 Loss of Coolant Accident.

SRO justification: The SRO must assess facility conditions (RCS pressure, temperature, Letdown flow, Charging flow, and facility MODE) and select the appropriate procedure for an abnormal/emergency situation.

Plausibility:

A. The examinee could recognize that a LOCA is in progress and think that in MODE 3 going to Attachment A and manually initiating SIAS would be appropriate.
B. Correct answer.
C. The examinee may think that a steam leak has occurred. This would not be correct as a steam leak would cause temperature to lower. Entering the AOP for Excessive Steam load would be appropriate for a steam leak.
D. The examinee may think that a steam leak has occurred. This would not be correct as a steam leak would cause temperature to lower. Entering the EOP for an Excessive Steam Demand Event would be appropriate for a steam leak.

Technical Reference(s): AOP 2568, Reactor Coolant System Leak, rev. 017

Provided reference(s): None

Learning Objective: (283280) ILT Given a set of plant conditions while following the instructions of AOP 2568, “Reactor Coolant Leak” and AOP 2568A, “RCS Leak MODE 4, 5, 6 and Defueled”, determine which procedure should be entered to respond to new conditions.

Source: ☐ New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

Comments: NRC feedback 7-28. Feedback was to rephrase the question to state with Pressurizer level and pressure lowering and RCS temperature stable, what does this indicate and what action will the US direct. And change answer to match the new question. Also to write SRO justification, make

Tc ($T_{COLD}$) the same throughout examine and remove periods from bullets. Change completed 8-5-22. Modified answers based on feedback from validator (A Leone). The answers were non-discriminating. DF 8/15/22

Question #85

The plant has experienced a Loss of Offsite Power (LOOP). The crew has entered EOP 2528, Loss of Offsite Power/Loss of Forced Circulation.

Conditions are as follows:

- $T_{HOT}$ = 537 °F and stable.
- $T_{COLD}$ = 535 °F and stable.
- $T_{CET}$ = 550 °F and rising.
- $P_{RCS}$ = 2250 psia and stable.
- PZR Level = 40 % and stable.

The US will direct steps to ___1.___ because ___2.___.

A. 1. ENSURE Single Phase Natural Circulation
   2. low RCS heat removal is causing a bubble to form in the head

B. 1. PERFORM Void Elimination actions
   2. voids forming in the fuel channels are restricting natural circulation flow

C. 1. ENSURE Single Phase Natural Circulation
   2. low RCS heat removal is restricting natural circulation flow

D. 1. PERFORM Void Elimination actions
   2. voids forming in the fuel channels are causing reverse natural circulation flow

Question #85 ☐ RO ☒ SRO

Tier # 1 Group # 2 K/A #E13 G2.1.20 Importance Rating: 4.6

K/A Statement: Loss of Forced Circulation and/or LOOP and/or a Blackout. Ability to interpret and execute procedure steps

Proposed Answer: **C**

Justification: The given conditions indicate some sort of restriction in natural circulation flow. CET temperatures and Hot Leg are not within 10 °F of each other. The operator would take action to increase the steaming rate to lower Cold Leg temperatures, restarting natural circulation flow.

Plausibility:

A. A bubble being formed in the head would cause pressurizer level to increase. Also, CET temperatures are greater than 30 degrees subcooled.
B. Voids forming can cause natural circulation to stall but the voids would manifest themselves as changes in pressurizer level and pressure.
C. Correct
D. Voids forming can cause natural circulation to stall but the voids would manifest themselves as changes in pressurizer level and pressure.

Technical Reference(s): EOP 2528, Loss of Offsite Power/Loss of Forced Circulation

Provided reference(s): None

Learning Objective: 283722 – Given a set of plant conditions concerning an Electrical Emergency, determine if natural circulation exists.

Source: Bank #415188.

History: 2011 NRC exam Q99

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

Comments: NRC feedback 7-28. Feedback was to change “experience” to “experienced”, add EOP 2528 to the stem, remove “you” from the question and rephrase, change “eliminate voids to “perform void elimination actions”, and change the second part of answer “D” for “is” causing to “are” causing. Change completed 8-5-22. Change 8-12-22 based on AL comments. Changed $T_{HOT}$, $T_{COLD}$, and CETs to in line with low RCS heat removal. Change completed 8-12-22.

Question #86

The plant is operating at 100% power.

The following events occur sequentially:

a) Annunciator RCP A UPPER SEAL PRES HI (C-02/3, CA-18) alarms.
b) Annunciator RCP A BLEED-OFF FLOW HI (C-02/3, BB-17) alarms.
c) Annunciator RCP A BLEED-OFF FLOW HI (C-02/3, BB-17) clears.
d) Annunciator RCP A BLEED-OFF FLOW LO (C-02/3, CA-17) alarms and remains lit.

1. This sequence of annunciators was the result of the…

   AND

2. What action must be directed?

A. 1. Controlled Bleed Off relief valve failing open.
   2. Trip the reactor and turbine, secure the "A" RCP and go to EOP 2525, "Standard Post Trip Actions".

B. 1. Controlled Bleed Off relief valve failing open.
   2. Enter AOP 2575, "Rapid Downpower", reduce power, remove the unit from service, and secure the "A" RCP.

C. 1. Controlled Bleed Off excess flow check valve seating.
   2. Enter AOP 2575, "Rapid Downpower", reduce power, remove the unit from service, and secure the "A" RCP.

D. 1. Controlled Bleed Off excess flow check valve seating.
   2. Trip the reactor and turbine, secure the "A" RCP and go to EOP 2525, "Standard Post Trip Actions".

Question #86 ☐ RO ☒ SRO

Tier # 2 Group # 1 K/A #003 A2.01 Importance Rating: 4.0

K/A Statement: Ability to (a) predict the impacts of the following on the Reactor Coolant Pump System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Problems with RCP seals, especially seal leakoff rates

Proposed Answer: D

Justification: The RCP A UPPER SEAL PRES HI annunciator is indicative of a failure of the "A" RCP middle or lower seal. This results in a high bleed-off flow through the "A" RCP seals resulting in a RCP A BLEED-OFF FLOW HI annunciator. At 10 gpm, the excess flow check valve will close causing the RCP A BLEED-OFF FLOW HI annunciator to clear and the RCP A BLEED-OFF FLOW LO to annunciate (0.75 gpm). At this point, the RCP seal package has no cooling flow and the RCP must be tripped. Procedurally, the reactor and turbine are tripped prior to tripping the affected RCP.

Plausibility:

A. The RCP bleed-off relief valve failed open would likely result in high bleed-off flow on all four (4) RCPs and not a low bleed-off flow annunciator. Additionally, "A" RCP seal pressures are dependent on seal conditions and not necessarily the flow path of bleed-off. Plausible because the examinee may not understand how the bleed-off relief valve would impact an individual RCP bleed-off flow and pressure. The examinee could also think each RCP has a bleed-off relief valve.
B. The individual indications provided could be a result of a failure of the "A" RCP middle and lower seals. However, the indications together would indicate a loss of bleed-off flow through the "A" RCP seals requiring the RCP to be stopped. If the lower and middle seals had failed, then the action is correct. Plausible because the Annunciator Response Procedures for the Upper Seal High Pressure and Bleed-off High Flow annunciators state that these alarms may be indicative of lower and middle seal failures.
C. A failed RCP vapor seal will also will also cause a RCP A BLEED-OFF FLOW LO annunciator alarm but not a RCP A BLEED-OFF FLOW HI. Plausible because a failed RCP vapor seal will also cause a RCP A BLEED-OFF FLOW LO annunciator alarm when all bleed-off flow is going into Containment, in which case AOP 2575 would be a procedure the operator could think is the correct action since it would lower reactor power prior to tripping to minimize the perturbation on the RCS parameters to prevent exacerbating the leak.
D. Correct answer

Technical Reference(s): AOP 2586, RCP Malfunctions, rev. 007
ARP 2590B-077, RCP A UPPER SEAL PRESS HI, rev. 002
ARP 2590B-069, RCP A BLEED-OFF FLOW LO, rev. 001
ARP 2590B-068, RCP A BLEED-OFF FLOW HI, rev. 001

Provided reference(s): None

Learning Objective: (286064) given a set of plant conditions determine if any RCP trip criteria are met.

Source: Bank

History: This is a bank question (452222) that was on the Millstone 2, 2016 NRC exam (Q76).

Answer “B was changed some because AOP 2586, RCP Malfunction replaced
OP 2301C for evaluating RCP malfunctions and middle seal was replaced with lower and middle seal to make the action more plausible.
Cognitive Level: Comprehension or Analysis
10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.
Comments: Accepted by the NRC 7-28-22.

Question #87

The plant is operating at 100% power.

- Calorimetric power = 2699.8 $MW_{TH}$ and rising.

Given these conditions, answer the following:

1. The power rise is due to RBCCW flow ___1.___ through the Letdown heat exchanger.

   AND

2. What procedure will the Unit Supervisor utilize to address this situation?

A. 1. increasing
   2. OP 2204, Load Changes

B. 1. decreasing
   2. OP 2204, Load Changes

C. 1. increasing
   2. AOP 2585, Immediate Operator Actions

D. 1. decreasing
   2. AOP 2585, Immediate Operator Actions

Question #87 ☐ RO ☒ SRO

Tier # 2 Group # 1 K/A #004 G2.1.37 Importance Rating: 4.6

K/A Statement: CVCS Chemical and Volume Control System. Knowledge of procedures, guidelines, or limitations associated with reactivity management

Proposed Answer: **A**

Justification: Increasing RBCCW flow through the Letdown Hx will add positive reactivity to the reactor. The increased RBCCW flow lowers Letdown temperature flowing into the primary demineralizers which, in turn, causes the ion bed to attract boron (strips boron from the RCS). OP 2204, Load Changes, directs the operator to reduce power to < 2699.8 $MW_{TH.}$

Plausibility:

A. Correct.
B. Decreased RBCCW flow will raise Letdown temperature flowing into the demineralizers causing boron to be transferred to the RCS. The examinee may think that lowering RBCCW flow may lower Letdown temperature, adding positive reactivity.
C. Increased RBCCW flow will add positive reactivity to the RCS. OP2585, Immediate Operator Actions, does not have a section to deal with overpower events associated with this type event.
D. Decreased RBCCW flow will add negative reactivity to the RCS. The examinee may think that lowering RBCCW flow may lower Letdown temperature, adding positive reactivity. OP2585 does not have a section to deal with this scenario.

Technical Reference(s): OP-AP-300, Reactivity Management
OP 2204, Load Changes

Provided reference(s): None

Learning Objective:284584 – Become familiar with OP-AP-300, Reactivity Management, so that the procedure becomes an effective tool to prevent unplanned reactivity changes.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(6): Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programing and determination of various internal and external effects on reactivity.

Comments: This was one of the NRC pre-submittal 10 questions. The NRC's comments were that we need to explain the plausibility for answer B. Not enough on why and examinee might think this. Is it a misconception? Not sure what someone is suppose too think about lowering RBCCW flow and it's effect on power. Re-wrote justification based on NRC comments. Modified question to read better. DF 6/14. NRC feedback 7-28. Feedback was to rephrase the question by inputting …given

RBCCW flow through the Letdown Heat Exchanger… , removing “reduce turbine load, and adding what procedure will address this situation.  Change completed 8-5-22.

Question #88

1. The Condensate Storage Tank (CST) is considered OPERABLE with a minimum contained volume of ___1.___ gallons.

   AND

2. The TS required ACTION if the CST minimum volume is NOT met, and fire water is not available as a backup, is ___2.___.

A. 1. 225,000 (90% indicated level)
   2. within 8 hours, restore water volume within the limit or be in COLD SHUTDOWN within the next 36 hours

B. 1. 225,000 (90% indicated level)
   2. within 4 hours, restore water volume within the limit or be in HOT SHUTDOWN within the next 12 hours

C. 1. 165,000 (66% indicated level)
   2. within 8 hours, restore water volume within the limit or be in COLD SHUTDOWN within the next 36 hours

D. 1. 165,000 (66% indicated level)
   2. within 4 hours, restore water volume within the limit or be in HOT SHUTDOWN within the next 12 hours

Question #88 ☐ RO ☒ SRO

Tier # 2 Group # 1 K/A #061 G2.2.42 Importance Rating: 4.6

K/A Statement: AFW Auxiliary/Emergency Feedwater System. Ability to recognize system parameters that are entry-level conditions for TS

Proposed Answer: **D**

Justification: The Condensate Storage Tank (CST) shall be OPERABLE with a minimum contained volume of 165,000 gallons (TS 3.7.1.3). The total tank volume is 250,000 gallons. The 165,000 gallons equates to 66% indicated on the control board. The required ACTION with less than 165,000 gallons in the condensate storage tank is to within 4 hours a. restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours. Since fire water is not available as a backup ACTION "a" is the required action.

Plausibility:

A. Incorrect: The value of 225,000 gallons is not correct. This is plausible because this is where the CST recirculation system loses suction. The action to within 8 hours, restore water volume within limit or be in COLD SHUTDOWN within the next 36 hours is also not correct. This is plausible because going to COLD SHUTDOWN is a fairly standard action in TSs and the examinee could think that going to this MODE is required when the auxiliary feedwater water source becomes INOPERABLE.
B. Incorrect: The value of 225,000 gallons is not correct. This is plausible because this is where the CST recirculation system loses suction. Within 4 hours restore the water volume to within the limit or be in HOT SHUTDOWN within the next 12 hours is correct.
C. Incorrect: The value of 165,000 gallons is correct. The action to within 8 hours, restore water volume within limit or be in COLD SHUTDOWN within the next 36 hours is not correct. This is plausible because going to COLD SHUTDOWN is a fairly standard action in TSs and the examinee could think that going to this MODE is required when the auxiliary feedwater water source becomes INOPERABLE.
D. Correct.

Technical Reference(s): MP2 Technical Specifications 3.7.1.3

Provided reference(s): None

Learning Objective: 282389 – State the conditions which constitute a violation of a Technical Specification Limiting Condition for Operation.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases

Comments: NRC feedback 7-28. Feedback was to change the second part of the question to ask what is the bases for the CST minimum level and change the second of the answers with two bases. Also revises justification and plausibility statements. This change makes it an SRO level question since it is TSs and their bases. Change completed 8-6-22. Changed after validation because validators selected the incorrect answer of "cooldown the RCS to allow SDC to be placed in service". Changed to "cooldown the RCS to COLD SHUTDOWN. Feedback 8-22-22 was that the question was not SRO level because it was TS bases information associated with above-the-line LCO information. Revised question to ask application of required action of TS section 3. Completed 8-23-22.

Question #89

The plant has experienced a Station Blackout (SBO). The Turbine Driven Auxiliary Feedwater pump is feeding both Steam Generators at 300 gpm each.

1. Millstone Unit 2 is analyzed to cope with a Station Blackout (SBO) for ___1.___.

   AND

2. A ___2.___ pump must be started, within one hour, to ensure the core will remain covered for the duration of the coping time.

A. 1. 8 hours
   2. HPSI

B. 1. 8 hours
   2. Charging

C. 1. 10 hours
   2. HPSI

D. 1. 10 hours
   2. Charging

Question #89 ☐ RO ☒ SRO

Tier # 2 Group # 1 K/A #063 G2.4.17 Importance Rating: 4.3

K/A Statement: DC Electrical Distribution System. Knowledge of emergency and abnormal operating procedures terms and definitions

Proposed Answer: B

Justification: Millstone Unit 2 has been analyzed to cope with a Station Blackout (SBO) for 8 hours provided a Charging pump is started within the first hour of the event. This question meets the K/A because cope (coping) is an emergency procedure term (2.4.17) and a loss of DC (063) would prevent the vital AC busses from being powered from their Emergency Diesel Generator, thus resulting in a SBO on the loss of offsite power. DC power is control power for Vital AC. Vital AC breakers will not function without Vital DC power. Vital DC power is also required to operate the Turbine Drive Auxiliary Feedwater (TDAFW) pump. And the TDAFW pump is the only available feedwater source in a SBO.

Plausibility:

A. Incorrect. A Charging pump is required to be started within one hour of a SBO. This is plausible because the leakage from the RCS in assumed to be 112 gpm (6,720 gallons lost in first hour). A HPSI pump would be expected to be used to recover inventory with this much inventory lost.
B. Correct
C. Incorrect. Millstone 2 is an 8 hour coping plant and the analysis has a Charging pump started in the first hour to meet the 8 hour coping time. This is plausible because, 10 hours is the time the Condensate Storage Tank (CST) inventory can maintain the plant in Hot Standby, with steam discharge to atmosphere (which would be the condition on a SBO). And a HPSI pump is plausible because of the amount of inventory assumed to be lost in a SBO.
D. Incorrect. Millstone 2 is an 8 hour coping plant. This is plausible because, 10 hours is the time the Condensate Storage Tank (CST) inventory can maintain the plant in Hot Standby, with steam discharge to atmosphere (which would be the condition on a SBO).

Technical Reference(s):

1. Technical Specification Bases 3.7.1.3 Condensate Storage Tank.
2. EOP 2530 Station Blackout Lesson Text, E30-01-C, VISION 55702, rev. 6/3, pages 1, 16, 19-21.

Provided reference(s): None

Learning Objective: 283760 – Outline and Explain the bases for the major actions in EOP 2530, Station Blackout.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10 CFR55.43(b)(1): Conditions and limitations in the facility license.
Comments: NRC feedback 7-28. Feedback was to remove “operating” from the stem and add additional detail to the justification, that DC power is control power for vital AC breakers and also is used to operate the TDAFW pump. Change completed 8-6-22.

Question #90

The plant was at 100% power when the Reactor was manually tripped.
Given the following plant parameters:

- Containment pressure = 8 psig and rising.
- Pressurizer pressure = 1180 psia and lowering.
- Pressurizer level = 10% and stable.
- RVLMS = 80%.
- RCS Hot Leg temperature = 450 °F and lowering.

1. How will the Pressurizer Pressure Control System (PPCS) respond?

   AND

2. What Emergency Operating Procedure (EOP) is required to be entered?

A. 1. The PPCS will turn Pressurizer heaters OFF.
   2. EOP 2532, Loss of Coolant Accident

B. 1. The PPCS will turn Pressurizer heaters OFF.
   2. EOP 2536, Excess Steam Demand Event

C. 1. The PPCS will turn Pressurizer heaters ON.
   2. EOP 2532, Loss of Coolant Accident

D. 1. The PPCS will turn Pressurizer heaters ON.
   2. EOP 2536, Excess Steam Demand Event

Question #90 ☐ RO ☒ SRO

Tier # 2 Group # 1 K/A #010 A2.07 Importance Rating: 4.0

K/A Statement: Ability to (a) predict the impacts of the following on the Pressurizer Pressure Control System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: ESFAS actuations

Proposed Answer: **B**

Justification: A Containment pressure rise is due to either a RCS leak and/or Main Steam leak into Containment. Both events will cause pressurizer pressure to lower. Initially the Pressurizer Pressure Control System (PPCS) will see pressure lowering resulting in a deviation from the PPCS setpoint. This will raise the output of the Pressurizer Proportional Heaters and eventually energize the Backup heaters at 2200 psia. But once Pressurizer Level lowers to 20% all heaters will turn off. Pressurizer level is lost (less than 20%) so all heaters are off. EOP 2536 will be entered due to subcooling is 115 °F. An Excess Steam Demand event (ESDE) is the only major accident in which Subcooling rises.

Plausibility:

A. Incorrect. The Pressurizer heaters OFF is correct, but EOP 2532 is not correct. With the given conditions (RCS pressure = 1180 psia (565 °F) and Th = 450 °F ) subcooling is 115 °F. An ESDE is the only event in which subcooling increases. Therefore, a LOCA is not correct. This is plausible because the examinee could reason it is a LOCA due to low Pressurizer pressure, RVLMS at 80%, and might not evaluate subcooling or calculate subcooling correctly.
B. Correct.
C. Incorrect. The Pressurizer heaters ON and EOP 2532 are not correct. This is plausible because the examinee could understand that lowering pressure would turn the heaters on but not remember the low Pressurizer level heater cutout. A LOCA is also plausible due to low Pressurizer pressure, RVLMS at 80%, and might not evaluate subcooling or calculate subcooling correctly.
D. Incorrect. The Pressurizer heaters ON is not correct. This is plausible because the examinee could understand that lowering pressure would turn the heaters on but not remember the low Pressurizer level heater cutout.

Technical Reference(s):

1. EOP 2541, Appendix 1 Diagnostic Flowchart, rev. 002-00
2. EOP 2541, Appendix 2, rev. 002
3. OP 2204, Load Changes, rev. 046, Attachment 3 Pressurizer Pressure Control Program, Attachment 4 Pressurizer Level Control Program.

Provided reference(s): Steam Tables

Learning Objective: 281901 - ILT Given the plant with a steam bubble in the pressurizer, and given a pressurizer pressure deviation from setpoint, describe the response of the Pressurizer Pressure Control System including setpoints.

Source: New

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

Comments: NRC feedback 7-28. Feedback was to change the second part of the question from “will you enter” to “is required to be entered”. Change completed 8-6-22.

Question #91

The plant is in MODE 6.

- A full core offload is in progress.
- The Upender is in the VERTICAL position in the Refuel Pool.
- An irradiated fuel assembly is in the Refuel Machine.
- The Refuel Machine is near the Upender.

Spent Fuel Pool level begins to lower at approximately 1 inch per minute.

The Refuel SRO, in accordance with AOP 2578, Loss of Refuel Pool and Spent Fuel Pool Level, will direct which one of the following?

A. Lower the fuel assembly into an open core location, ensure the Upender is VERTICAL in the refuel pool, then close the Transfer Tube Isolation Valve.

B. Lower the fuel assembly into the south saddle and place the Upender in HORIZONTAL, do NOT close the Transfer Tube Isolation Valve.

C. Place the fuel assembly in the Upender and place Upender in HORIZONTAL, do NOT close the Transfer Tube Isolation Valve.

D. Place the fuel assembly in the Upender and transfer fuel assembly to the spent fuel pool, then close the Transfer Tube Isolation Valve.

Question #91 ☐ RO ☒ SRO

Tier # 2 Group # 2 K/A #033 A2.03 Importance Rating: 3.7

K/A Statement: Ability to (a) predict the impacts of the following on the Spent Fuel Pool Cooling System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Abnormal water level

Proposed Answer: **D**

Justification: AOP 2578 step 4.1.13 directs placing fuel in Upender, postioning to HORIZONTAL, transferring to the SFP, and then closing the Transfer Tube Isolation Valve. This question is SRO only as it requires assessing plant conditions (rate of RFP level decrease, location of leak in SFP) and then selecting a procedure or section of a procedure to mitigate, recover or with which to proceed. The question relates to duties of the Refueling SRO and cannot be answered by solely knowing systems knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigation strategy of a procedure.

Plausibility:

A. Incorrect. Per Step 4.1.13 the fuel assembly will be transferred to the Spent Fuel Pool (SFP). The Transfer Tube Isolation Valve would not be closed with the upender in containment. This is plausible because the core is listed as a safe location in the note preceding Step 4.1.11. The note does not list the upender as a possible transfer location. Examinee may select this answer if they know the guidance in the note regarding Refueling SRO tasks but does not know the procedural guidance that follows the note.
B. Incorrect. Closing the transfer tube isolation valve is a priority to limit the loss of level to the affected pool. Step 4.1.13 directs transfer to the SFP. This is plausible because the examinee may reason that leaving the transfer tube gate open is preferable because the rate of SFP level drop will significantly increase with the gate closed and closing the gate limits capability for maintaining SFP level through use of LPSI injection into the core.
C. Incorrect. Leaving the assembly in the upender, even in HORIZONTAL is not a designated safe storage location. Closing the transfer tube isolation valve is a priority to limit the loss of level to the affected pool. This is plausible because the examinee may reason that leaving the transfer tube gate open is preferable because the rate of SFP level drop will significantly increase with the gate closed and closing the gate limits capability for maintaining SFP level through use of LPSI injection into the core.
D. Correct

Technical Reference(s):

AOP 2578, Loss of Refuel Pool and Spent Fuel Pool Level, Rev. 008-00, pages 2, 3, and 9.

Provided reference(s): None

Learning Objective: 283463 –Describe the general sequence of operations for AOP 2578 “Loss of Refuel Pool and Spent Fuel Pool Level.

Source: Bank

History: 451946, NRC 2014 Question 92

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: NRC feedback 7-28. Feedback was to remove double periods from answer "A" and "C". Change completed 8-6-22.

Question #92

The plant is operating at 100% power when Vital DC bus DV-10 is lost.

Plant conditions are as follows:

- Pressurizer pressure = 2030 psia and lowering.
- Pressurizer level = 35% and lowering.
- Steam Generator pressure = 905 psia and lowering.
- Annunciator STM GEN NO. 1 SAFETY RELIEF VALVE OPEN (C-05, DA-17) is lit.

1. What procedure will the Unit Supervisor use to direct action to address the open safety valve?

   AND

2. What action will the Unit Supervisor direct?

A. 1. EOP 2525, Millstone Unit 2 Standard Post Trip Actions
   2. Lower the Turbine Bypass valve setpoint to 840 psig.

B. 1. EOP 2525, Millstone Unit 2 Standard Post Trip Actions
   2. Lower the Atmospheric Dump Valve setpoint to 880 psia.

C. 1. EOP 2541, Appendix 4, Millstone Unit 2 Followup Actions
   2. Lower the Turbine Bypass valve setpoint to 840 psig.

D. 1. EOP 2541, Appendix 4, Millstone Unit 2 Followup Actions
   2. Lower the Atmospheric Dump Valve setpoint to 880 psia.

Question #92 ☐ RO ☒ SRO

Tier # 2 Group # 2 K/A #035 A2.02 Importance Rating: 4.2

K/A Statement: Ability to (a) predict the impacts of the following on the Steam Generator System and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operations: Reactor trip/turbine trip

Proposed Answer: B

Justification: EOP 2525, Millstone Unit 2 Standard Post Trip Actions and lower the Atmospheric Dump Valve setpoint to 880 psia is correct.
OP 2260, Unit 2 EOP User's Guide, specifies that "due to current MSSV blowdown setpoints (~ 880 psia), it is possible for MSSV(s) to be open post-trip when the steam dumps are not available AND have SG pressure within the normal control band. To assess if a MSSV is actually stuck open, it is recommended the BOP adjust ADV automatic setpoint(s) to the lower end of the control band."

Plausibility:

A. Lowering the Turbine Bypass valve setpoint to 840 psig would lower Steam Generator pressure if the MSIVs are open. But both MSIVs closed when DV-10 was lost. The student may not remember that the MSIVs close on a loss of DV-10
B. Correct Answer.
C. The student may not remember the administrative guidance for adjusting setpoint on the ADV in an attempt to close the MSSV is applied to EOP 2525. And since it is not a specific step in EOP 2525 they may think the guidance should be applied when in EOP 2541 Appendix 4. And the students may not remember DV-10 closes both MSIVs thereby making the Turbine Bypass valve unavailable.
D. The student may not remember the administrative guidance for adjusting setpoint on the ADV in an attempt to close the MSSV is applied to EOP 2525. And since it is not a specific step in EOP 2525 they may think the guidance should be applied when in EOP 2541 Appendix 4. Lowering the Atmospheric Dump Valve setpoint to 880 psia is correct.

Technical Reference(s): OP 2260, Unit 2 EOP User's Guide, revision 017

Provided reference(s): None

Learning Objective: (283650) ILT Given a set of plant conditions, determine the action required in the accordance with immediate actions of EOP 2525, Standard Post Trip Actions.

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: NRC feedback 7-28. Feedback was to change the question to ask the procedure question first and action second (flip each answer), change answers to match, and remove "you" from the question. Change completed 8-6-22.

Question #93

The plant is in MODE 6; with fuel movement is in progress.

- Containment Purge is in service.
- Containment Atmosphere Radiation Monitors, RM-8123/8262 are out of service.

In accordance with ___1.___, a ___2.___ operator is assigned to close the Containment Purge Valves (AC-4, 5, 6, and 7) to meet administrative requirements when moving irradiated fuel assemblies within containment.

A. 1. TS 3.6.3.2, CONTAINMENT VENTILATION SYSTEM
   2. Dedicated (sole responsibility)

B. 1. TS 3.6.3.2, CONTAINMENT VENTILATION SYSTEM
   2. Designated (has concurrent duties)

C. 1. TS 3.9.4, CONTAIMENT PENETRATIONS
   2. Dedicated (sole responsibility)

D. 1. TS 3.9.4, CONTAIMENT PENETRATIONS
   2. Designated (has concurrent duties)

Question #93 ☐ RO ☒ SRO

Tier # 2 Group # 2 K/A #072 G2.1.35 Importance Rating: 3.9

K/A Statement: Area Radiation Monitoring System. Knowledge of the fuel handling responsibilities of SROs

Proposed Answer: D

Justification: TS 3.9.4, CONTAIMENT PENETRATIONS, requires that "Each penetration providing direct access from the containment atmosphere to the outside atmosphere (i.e., Containment Purge Valves) shall be either: closed by automatic isolation or be capable of being closed under administrative control." Since the CTMT radiation monitors are secured and the purge valves open, a designated person is required to be available to provide isolation of containment from the outside atmosphere during movement of irradiated fuel assemblies within containment.

Plausibility:

A. Incorrect. LCO 3.6.3.2, CONTAINMENT VENTILATION SYSTEM is not correct. This TS is applicable in MODES 1 – 4 and not in MODE 6. A dedicated operator (sole responsibility) is used to maintain a OPERABILITY of a system or component and this is also not correct. This is plausible because the examinee may think the Purge valve closure requirements of LCO 3.6.3.1 are required during the movement of irradiated fuel assemblies in MODE 6.
B. Incorrect. LCO 3.9.4, CONTAIMENT PENETRATIONS is correct. A dedicated operator (sole responsibility) is used to maintain a OPERABILITY of a system or component is not correct. This is plausible because the examinee may think the Containment Purge Isolation (CPVIS) is required to be OPERABLE to achieve containment closure requirements in MODE 6.
C. Incorrect. LCO 3.6.3.2, CONTAINMENT VENTILATION SYSTEM is not correct. This TS is applicable in MODES 1 – 4 and not in MODE 6. A designated operator is required to close the CTMT Purge valves is correct. This is plausible because the examinee may think the Purge valve closure requirements of LCO 3.6.3.1 are required during the movement of irradiated fuel assemblies in MODE 6.
D. Correct.

Technical Reference(s): Unit 2 Technical Specifications LCO 3.9.4, CONTAIMENT PENETRATIONS.

Provided reference(s): None

Learning Objective: 283455 – Given a set of plant conditions, determine Technical Specification applicability during a fuel handling accident

Source: New

History:

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(7): Fuel handling facilities and procedures.
Comments: Rewrote question due to feedback from C. Carpenter. DF 6/14. NRC feedback 7-28. Feedback was to change the question to ask what Technical Specification first and then ask the type of operator second (flip the question and answers). Changed plausibility statements to align with reorder of question. Change completed 8-6-22.

Question #94

A Large Break Loss of Coolant Accident occurred 15 hours ago.

The following are the current plant conditions:

- RWST level = 8% and stable.
- RCS Pressure = 35 psia and lowering very slowly.
- Containment Pressure = 20 psig and lowering very slowly.
- Safety Injection flow on the PPC and Main Control Boards = 360 gpm.
- VR-11 is de-energized.

Given EOP 2541, Appendix 2, Figures 3, 4, and 5, determine whether Safety Injection flow is acceptable and the appropriate procedure to address the conditions.

1. Safety Injection flow ____1.____ for the existing conditions.

   AND

2. Procedure ____2.____ will address the conditions.

A. 1. is NOT acceptable
   2. EOP 2532, Loss of Coolant Accident

B. 1. is NOT acceptable
   2. EOP 2540, Functional Recovery

C. 1. is acceptable
   2. EOP 2532, Loss of Coolant Accident

D. 1. is acceptable
   2. EOP 2540, Functional Recovery

Question #94 ☐ RO ☒ SRO

Tier # 3 Group # 1 K/A #2.1.25 Importance Rating: 4.2

K/A Statement: Ability to interpret reference materials, such as graphs, curves, and tables (reference potential)

Proposed Answer: **C**

Justification: The examinee must make an assessment that SRAS has actuated. There are three Safety Injection curves in the figures provided. They must select Figure 4 – Post - SRAS Minimum Required SI Flow. Then understand that with VR-11 de-energized that indicated flow (300 gpm) must be doubled (600 gpm). Using Figure 4 the 600 gpm of Safety Injection flow meets the Safety Injection flow requirements. EOP 2532 Loss of Coolant accident is the appropriate procedure to address the conditions.

Plausibility:

A. Incorrect. Safety Injection (SI) flow is acceptable. This is plausible because the examinee could use the wrong curve (Figure 3 – Pre – SRAS Minimum Required SI Flow) and using this curve SI flow would not be acceptable. Or use of the correct curve and not taking into account the de-energization of VR-11 SI flow would not be acceptable. EOP 2532 has steps to raise SI flow if the SI flow curve is not acceptable.
B. Incorrect. Safety Injection (SI) flow is acceptable. This is plausible because the examinee could use the wrong curve (Figure 3 – Pre – SRAS Minimum Required SI Flow) and using this curve SI flow would not be acceptable. Or use of the correct curve and not taking into account the de-energization of VR-11 SI flow would not be acceptable. EOP 2540 is the procedure which is entered if the safety functions are not satisfied in the procedure you are currently in.
C. Correct.
D. Incorrect. EOP 2540, Function Recovery is not correct since SI flow is being acceptable. This is plausible since they may think that 20 psig in Containment is too low and EOP 2532 will not address a loss of Containment Integrity. But Containment pressure would be lowered by spray and CARs and this is equal to RCS pressure (35 psia versus 20 psig).

Technical Reference(s): EOP 2541, Appendix 2, Figures, rev. 002

Provided reference(s): EOP 2541, Appendix 2, Figures 3, 4, and 5, rev. 002.

Learning Objective: 283817 - SRO evaluate plant conditions and direct performance of instructions and contingency actions in accordance with EOP 2532, Loss of Primary Coolant.

Source: New

History: Written from 286500 and 406205.

Cognitive Level: Comprehension or Analysis

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

Comments: NRC feedback 7-28. Feedback was to change “met” to “acceptable” in part one of each answer. Changed “met” in plausibility statements to “acceptable”. Change completed 8-6-22.

Question #95

Why must the Reactor be subcritical for a minimum of 100 hours prior to moving irradiated fuel?

A. To allow sufficient radioactive decay of short-lived fission products.

B. To allow sufficient time for the radioactivity decay of N-16.

C. To ensure decay heat will not exceed the cooling capacity of the SFPC system.

D. To ensure that off-loading the core to the SFP will not result in criticality.

Question #95 ☐ RO ☒ SRO

Tier # 3 Group # 1 K/A #2.1.40 Importance Rating: 3.9
K/A Statement: Knowledge of refueling administrative requirements

Proposed Answer: A
Justification: The bases for the requirement for the reactor being subcritical for a minimum of 100 hours prior to movement of irradiated fuel is to ensure that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products so the calculated radiological dose consequences of the fuel handling accident are bounding. Reference TS 3.9.3.1.
Plausibility:

A. Correct.
B. Incorrect. Nitrogen -16 is not correct. This is plausible because fission produces N-16 and the examinee could understand that N-16 lowers as power lowers and reason that some time is needed to have it fully decay away.
C. Incorrect. To ensure decay heat will not exceed the cooling capacity of the SFPC system is not correct. This is plausible because this is the bases for the reactor required to be in MODE 5, 6 or core defueled until either 616 hours has passed since sub-critical or heat load in the SFP is less than 10.16 X10E6 BTU/hr. Reference TRM 3.9.3.3.
D. Incorrect. To ensure that off-loading the core in the SFP will not result in criticality is not correct. This is plausible because the examinee could reason that time is required for the reactivity of the fuel, being removed, to decrease to prevent criticality in the SFP. This is actually the bases for the boron requirements in the SFP. Reference TS 3.9.1.

Technical Reference(s): TS 3.9.3.1, Refueling Operations – Decay Time.
Provided reference(s): None
Learning Objective: 282688 – SRO Explain the bases for all Technical Specifications and (TRM) Technical Requirements that apply during Refueling Operations.
Source Bank #414120:
History:
Cognitive Level: Memory or Fundamental Knowledge
10CFR55.43(b)(2): Facility operating limitations in the technical specifications and their bases.
Comments: Accepted by the NRC 7-28-22.

Question #96

Annunciator “HEATER DRAIN PUMP TROUBLE” has become a nuisance alarm and will be removed from service.

1. Approval is required from the ___1.___ to remove this annunciator from service.

   AND

2. A 10CFR50.59 Applicability Review ___2.___ required for this activity.

A. 1. Shift Manager
   2. IS

B. 1. Shift Manager
   2. IS NOT

C. 1. Engineering Manager
   2. IS

D. 1. Engineering Manager
   2. IS NOT

Question #96 ☐ RO ☒ SRO

Tier # 3 Group # 2 K/A #2.2.5 Importance Rating: 3.2

K/A Statement: Knowledge of the process for making design or operating changes to the facility, such as 10 CFR 50.59, "Changes, Tests and Experiments," screening and evaluation processes, administrative processes for temporary modifications, disabling annunciators, or installation of temporary equipment

Proposed Answer: A

Justification: The Shift Manger authorizes removal of annunciators. This is detailed in OP 2387A, "Annunciator System Operation and Control", step 4.3.12. A 10CFR50.59 Applicability Review is required per step 4.3.6 of OP 2387A.

Plausibility:

A. Correct Answer
B. Incorrect. This is plausible because the student could think a 10CFR50.59 Applicability Review is not required because the Heater Drain system is not Safety Related. It is not powered by a vital source nor is it required to mitigate the consequences of a design basis accident.
C. Incorrect. This is plausible because the student could think that removal of an annunciator is a temporary modification and requires Engineering Manager's approval. Temporary modifications fall under designs and are controlled through Engineering. A 10CFR50.59 Applicability Review is required, is correct.
D. Incorrect. This is plausible because the student could think that removal of an annunciator is a temporary modification and requires Engineering Manager's approval. Temporary modifications fall under designs and are controlled through Engineering. The student could also think a 10CFR50.59 Applicability Review is not required because the Heater Drain system is not Safety Related. It is not powered by a vital source nor is it required to mitigate the consequences of a design basis accident.

Technical Reference: OP 2387A, "Annunciator System Operation and Control", revision 007-03.

Provided reference(s): None

Learning Objective:

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(3): Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments: This was one of the (10) NRC pre-submittal questions and the NRC's comment was on possibly removing "Design" from answers and enhancing justification. JWR on 6/16 changed Design Engineering Manager to Engineering Manager in answers and enhanced justification. Accepted by the NRC 7-28-22.

Question #97

The plant is operating at 100% power.

A troubleshooting plan was written to investigate leak-by of 2-SI-217, Loop 1A Non-Return Check Valve.

- The plan requires filling the #1 SIT.
- The SIT will remain OPERABLE during the performance of the troubleshooting plan.

Given MA-AA-103, Conduct of Troubleshooting, Attachment 1, Risk and Rigor Determination Matrix, what level of risk would be assigned to the plan?

A. No risk

B. Low

C. Medium

D. High

Question #97 ☐ RO ☒ SRO

Tier # 3 Group # 2 K/A #2.2.20 Importance Rating: 3.8
K/A Statement: Knowledge of the process for managing troubleshooting activities

Proposed Answer: B
Justification: The definition of Low risk from MA-AA-103 (attachment 1, page 35) is the equipment is not removed from service and troubleshooting could have an effect on plant equipment, but should not present a risk of causing an undesirable plant transient, reactivity change or reportable event, and no unusual personnel safety risk.
Plausibility:

A. The equipment is not removed from service. A 'No risk' designation requires the equipment to be out of service. The examinee may think that since the check valve is not in service (no flow through it), there is no risk
B. Correct.
C. Incorrect. There is no risk in causing an undesirable plant transient. The examinee may think that there is a possibility of affecting reactivity when filling the SIT since the check valve is not seating properly. Filling SITs will not affect reactivity.
D. Incorrect. There is no risk for a major plant transient. The examinee may feel that, due to the importance of the SI system, that a high risk designation may be the conservative choice.

Technical Reference(s): MA-AA-103, Conduct of Troubleshooting
Provided reference(s): MA-AA-103, Conduct of Troubleshooting; Attachment 1, Risk and Rigor Determination Matrix
Learning Objective: 217665 – Discuss the considerations involved in approving a Trouble Shooting Plan.
Source: New
History:
Cognitive Level: Comprehension or Analysis
10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.
Comments: NRC feedback 7-28. Feedback was to find a reason and adjust justification. Added reason to the stem (to investigate leak-by of a check valve) and a justification changes was not needed because the check valve was already in the justification. Change completed 8-8-22.

Question #98

The plant has experienced a S/G tube leak.

In accordance with AOP 2569, Steam Generator Tube Leak:

1. Why is blowdown re-established on the affected S/G?

   AND

2. How is Blowdown from the affected S/G aligned?

A. 1. To maintain secondary chemical concentrations in the S/G.
   2. Blowdown Tank draining to the Seal Well.

B. 1. To minimize the concentration of activity in the S/G.
   2. Blowdown Tank draining to the Seal Well.

C. 1. To maintain secondary chemical concentrations in the S/G.
   2. Blowdown Quench Tank draining to Clean Liquid Rad Waste.

D. 1. To minimize the concentration of activity in the S/G.
   2. Blowdown Quench Tank draining to Clean Liquid Rad Waste.

Question #98 ☐ RO ☒ SRO

Tier # 3 Group # 2 K/A #2.3.11 Importance Rating: 4.3

K/A Statement: Ability to control radiation releases

Proposed Answer: **D**

Justification: If the plant is to stay on line during a tube leak, S/G B/D is placed in service in order to reduce the concentration of activity in the S/G. The B/D is aligned to the Quench Tank since radioactive discharges are not permitted through the B/D Tank to the Seal Well.

Plausibility:

A. B/D is aligned to reduce radioactive activity concentrations. The Blow Tank is not permitted for radioactive discharges.
B. The Blow Tank is not permitted for radioactive discharges.
C. B/D is aligned to reduce radioactive activity concentrations.
D. Correct

SRO justification: The SRO must assess facility conditions and select the appropriate procedure for an abnormal/emergency situation.

Technical Reference(s): AOP 2569, Steam Generator Tube Leak

Provided reference(s): None

Learning Objective: 283307 – Describe the general sequence of operations for AOP 2569, Steam Generator Tube Leak.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(4): Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination situations.

Comments: NRC feedback 7-28. Feedback was to revise stem and answers “A” and “C”. Change stem to read the plant has experienced a SGTL, change questions to reference procedure AOP 2569, remove “you” from questions, and replace “reduce” with “maintain” in answers “A” and “C”. Change completed 8-8-22. Modified stem based on validator feedback (A Leone). Removed bullets since they added no value to the question. DF 8/15/22

Question #99

The plant has declared a SITE AREA EMERGENCY.

- The Emergency Operating Facility (EOF) has been activated.
- The EOF Director of Station Emergency Operations (DSEO) has assumed control.
- The Shift Manager has entered MP-26-EPI-FAP01-002, Manager of Control Room Operations (MCRO).

Given the conditions above, what is the Shift Manager responsible for?

A. Declaring and making notifications of Emergency Action Level Classification changes.

B. Developing Protective Action Recommendations that result from changes in plant and notifying State DEEP.

C. Authorizing exposure upgrades, up to 25 Rem, for on-shift Operations personnel.

D. Coordinating with the Assistance Director of Technical Support to establish minimum equipment isolation.

Question #99 ☐ RO ☒ SRO

Tier # 3 Group # 3 K/A #2.4.40 Importance Rating: 4.5
K/A Statement: Knowledge of SRO responsibilities in emergency plan implementing procedures (SRO Only).

Proposed Answer: D
Justification: Coordinating with the Assistant Director of Technical Support (ADTS) to establish minimum equipment isolation. This is step 2 of MP-26-EPI-FAP01-002, Manager of Control Room Operations (MCRO) procedure.
Plausibility:

A. Incorrect. Declaring and making notifications of Emergency Action Level (EAL) Classifications changes is not correct. The Shift Manager evaluates plant conditions and makes recommendations to the ADTS once the EOF DSEO has activated the EOF and relieved the CRDSEO. This is step 5 of MP-26-EPI-FAP01-002, Manager of Control Room Operations (MCRO) procedure. This is plausible because the CRDSEO does have this responsibility until relieved by the EOF DSEO.
B. Incorrect. Developing Protective Action Recommendations (PAR) that result from changes in plant and notifying State DEEP is not correct. The EOF DSEO has the responsibility for PARs once the Emergency Plan is activated and the EOF DSEO has assumed control of the Emergency Organization. This is plausible because the CRDSEO does have this responsibility until relieved by the EOF DSEO.
C. Incorrect. Authorizing exposure upgrades, up to 25 Rem, for on-shift Operations personal is not correct. Once the Emergency Organization is activated Assistance Director approval is required for exposure upgrades up to 25 Rem. This is plausible because an examinee could reason that the Shift Manager still has the Operations on-shift under his direction throughout the emergency.
D. Correct.

Technical Reference(s):

1. MP-26-EPI-FAP01-002, Manager of Control Room Operations (MCRO), rev. 007, page 1.
2. MP-26-EPI-FAP01-001, Control Room – Director of Station Emergency Operations (CR-DSEO), rev. 017, pages 1, 2, and 17.
3. MP-26-EPI-FAP09, Radiation Exposure Controls, rev. 007, page 13.

Provided reference(s): None
Learning Objective:
Source: New
History:
Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: This was one of the (10) NRC pre-submittal questions. The NRC's comment was that this K/A should be replaced since the K/A is not an appropriate K/A for a SRO level question. New K/A, G2.4.40 was received from the NRC and a new question was written. NRC feedback 7-28. Feedback was to change stem to a "Site Area Emergency", change "personal" to "personnel" in answer "C", and spell out ADTS. Also spelled out EALs and PARs in answers to make answers of similar lengths. Change completed 8-8-22.

Question #100

The plant has experienced a Station Blackout. EOP 2530, Station Blackout, is in progress.

- CET temperatures are 1300 °F and increasing.

What is the procedurally directed action?

A. Use the Diagnostic Flowchart to re-diagnose the event and implement Once-Through-Cooling.

B. Transition directly to the Functional Recovery Procedures and implement Once-Through-Cooling.

C. Stay in EOP 2530, Station Blackout and continue restoration of power.

D. Go To SAMG 25 SAG-01, Main Control Room SAMG Initial Response.

Question #100 ☐ RO ☒ SRO

Tier # 3 Group # 4 K/A #2.4.51 Importance Rating: 4.0

K/A Statement: Knowledge of emergency operating procedure exit conditions (e.g., emergency condition no longer exists or severe accident guideline entry is required)

Proposed Answer: **D**

Justification: EOP 2530, Station Blackout, directs the operator to Go To SAMG 25 SAG-01, Main Control Room SAMG Initial Response, if CET temperatures exceed 1200 °F.

Plausibility:

A. 1300 °F constitutes a SFSC Acceptance Criteria not being met. The EOP User's Guide then gives the operator an opportunity to re-diagnose the event. The student may think initiating OTC will alleviate the condition but there are no HPSI pumps available.
B. The operator may believe the 1300 °F constitutes a SFSC Acceptance Criteria not being met. The EOP User's Guide then gives the operator an opportunity to Transition directly to the Functional Recovery Procedures.
C. The operator may reason that staying in the Station Blackout procedure is the appropriate course of action because it will restore power. And restoring power must be done before other safety functions can be addressed.
D. Correct

Technical Reference(s): EOP 2530, Station Blackout, rev. 019, page 19, step 21.

Provided reference(s): None

Learning Objective: 283760 - Outline and explain the bases for the major actions in EOP 2530, Station Blackout.

Source: New

History:

Cognitive Level: Memory or Fundamental Knowledge

10CFR55.43(b)(5): Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments: NRC feedback 7-28. Feedback was to change stem of question to "what is the procedurally directed action". Change completed 8-8-22.