>>QUESTION 1

The plant tripped from 100% power. EOP 2525, Standard Post Trip Actions, is in progress.

What instrumentation is used to determine Reactor Power five (5) minutes after the trip?

A. Calorimetric power.

B. Power Range power meters.

C. Wide Range Power meters.

D. CEAPDS Q-power.

<<

>> QUESTION 1

K&A Rating: 007EA2.01 (4.1)

K&A Statement: EPE 007 Reactor Trip, EA2.01 Ability to determine or interpret the following as they apply to a reactor trip: Decreasing power level, from available indications.

Key Answer: **C**

Justification:

A. **Incorrect:** Wide Range power meters are correct because they indicate from 10-8% up to 100%. The Calorimetric indicates power primarily with mass flow and enthalpy across the Steam Generators when the reactor is in the percent power range and will not be an accurate indication of actual reactor power.
Plausible: When operating at steady state higher power level, with Feedwater Flow not changing significantly, the Calorimetric is the most accurate indication of power.

B. **Incorrect:** Power range power indication is not correct because it's range is 0% to 125% and reactor power will be in the in the 10-1% to 10-4% range, Also Power range indications includes heat from decay heat and RCPs because they indicate the higher of Delta T and NI power.
Plausible: Candidate should know that Delta T power is the most accurate indication when power is changing. This is a note in AOP 2575, Rapid Downpower. Also Delta T power is affected by decay heat and RCPs, and this will cause indication on the Power range meters to be much higher than nuclear power.

C. **CORRECT:** Wide Range power meters are correct because they indicate from 10-8% up to 100%. And power after 5 minutes will be in the 10-1% to 10-3% range. Information contained in EOP 2525 SPTA does not specify the power indication to use. The candidate must understand the power indications available and their limitations.

D. **Incorrect:** CEAPDS power indicated is the higher of Delta T and NI power. Delta T will be indicated and NI will be more accurate because Delta T power includes heat from decay heat and RCPs.
Plausible: Indication available on CO4 which is in the same place you check CEA insertion.

References: EOP 2525 Rev 028-00, NIS-01-C Rev 5 Chg 2, None Student Ref: NONE

Learning Objective: 283649 (05424-MB) Outline the Instructions and Contingency Actions for the Immediate Actions in EOP 2525, Standard Post Trip Actions.

Question Source: new

Question History: new

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7

Comments (Question 1): 08/6/2020 jwr. Updated to incorporate the NRC comments from 7/24/2020. Their comments were to remove “most” from the stem, verify what is used by procedure, and ensure justification reflects answers. Removed “most” from stem. There are no reference materials that specify which power indication to use. Changed justification to state that information contained in EOP 2525 SPTA does not specify the power indication to use and that the candidate must understand the power indications available and their limitations.
05/31/20 jwr. PF comment to use CEAPDS Q power as answer “D” instead of CEAPDS monitor power. Changed answer “D” to address comment.
<<

>>QUESTION 2

Thirty minutes after a plant transient which caused the Pressurizer Safeties to lift, the following conditions exist:

- RCS is at a saturated condition.
- RCS pressure is 1000 psia .
- The Quench Tank Rupture Disc has blown.
- Containment pressure is 25 psig.

What Pressurizer Safety Valve discharge temperature would indicate that fluid is still leaking by the Safeties?

A. 240 °F.

B. 312 °F.

C. 348 °F.

D. 545 °F.

<<

>>QUESTION 2

K&A Rating: 008AK2.02 (2.7)

K&A Statement: Pressurizer (PZR) Vapor Space Accident. Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: Sensors and detectors

Key Answer: **B**

Justification:

A. **Incorrect:** 240 ˚F is the temperature for 25 psia (10 psig) instead of 40 psia (25 psig + 15 psi).
Plausible: It is reasonable for the candidate to not convert psig to psia to obtain the correct temperature. And to follow the Saturation Line down to 25 psia instead of straight across, which is which is correct, for a constant enthalpy process.

B. **CORRECT:** The open safety valve constitutes an isenthalpic (constant enthalpy) process. From the Mollier Diagram the temperature for 1000 psia expanded to 40 psia (25 psig + 15 = 40 psia) is 312 ˚F. This is obtained by following the 1000 psia constant pressure line down to The Saturation line. Then straight across (constant enthalpy) to the 40 psia constant pressure line.

C. **Incorrect**: 348 ℉ is the temperature of 1000 psia expanded down to the Saturation Line and to then expanded straight across (constant enthalpy) but stopping at the other side of the Saturation Line, instead of going further to 40 psia.
Plausible: This is similar to how the temperature is determined if pressure starts higher and is expanded to the end pressure below the Saturation Line. In this case, the end point constant pressure line is followed up to the Saturation Line to determine the correct temperature.

D. **Incorrect:** 545 ℉ is the saturation temperature for 1000 psia.
Plausible: This is a common misconception. That the fluid with be at a saturation condition, when in fact it is superheated.

References: Mollier Diagram, GFES Thermodynamic Processes Rev 0 November 2014
Student Ref: Steam tables, including Mollier Diagram

Learning Objective: 281978 (MB-03036) Describe the indications available to detect a leaking code safety valve.

Question Source: 56212 (310929)

Question History: Modified. Changed stem from 1200 psia to 1000 psia. And changed correct answer and two of other answers. Used NRC GFES Bank question P4040 as a reference.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7 / 45.7

Comments (Question 2): 08/06/2020 jwr. Completed changes to address the NRC's comments from their 07/24/2020 feedback. Their comments were to add a comma in the stem after "lift" and specify Steam Tables with a Mollier Diagram as a student reference.<<

>>QUESTION 3

The crew is performing EOP 2532, Loss of Coolant Accident.

- RBCCW SURGE TK LEVEL HI/LO (C-06/7, A-8) alarm is locked in.
- RBCCW Surge Tank level is reading 100%.

What action is directed by EOP 2532 for this condition?

A. VERIFY “RBCCW SURGE TK MAKEUP, RB-215”, is closed.

B. ENSURE PMW Pumps, P22A and P22B, are secured.

C. ENSURE CIAS has actuated and the RBCCW CTMT isolation valves are closed.

D. STOP any operating RCPs and close the RBCCW CTMT isolation valves.

<<

>>QUESTION 3

K&A Rating: 2.4.50 (RO 4.2 / SRO 4.0)

K&A Statement: EPE 009 Small Break LOCA G2.4.50 Emergency Procedures/Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Key Answer: **D**

Justification:

A. **Incorrect:** these actions are for RBCCW SURGE TK LEVEL HI/LO ARP
Plausible: might think the ARP should be used (**alarm comes in 91.4%)**

B. **Incorrect:** The PMW pumps are not secured
Plausible: May think securing the PMW will stop filling the RBCCW S/T if the valves fail open

C. **Incorrect**: CIAS won't isolate RBCCW loads in CTMT. EOP 2532 checks for LOCA outside CTMT by RMs alarming and ENSURING CIAS actuated
Plausible: might think CIAS will isolate some RBCCW loads in CTMT

D. **CORRECT:** EOP 2532 RNO for RBCCW Surge tank NOT rising is to stop operating R CPs and isolate RBCCW to CTMT

References: EOP 2532, LOCA
ARP 2590E-045 Rev 1
ARP 2590E-046 Rev 0

Student Ref: NONE

Learning Objective: 283789 (MB-05940) Outline and explain the bases for the major actions in EOP 2532, Loss of Coolant Accident

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X

Comprehensive/Analysis:

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

Comments (Question 3): 6/10/20, modify answers C & D. Added RBCCW S/T Hi/Lo alarm in stem, changed answer 'B' to more plausible answer (NRC comments) DF 7/27/20

<<

>>QUESTION 4

The plant is at 45% power. A loss of Reactor Building Closed Cooling Water (RBCCW) event is in progress.

Which of the following conditions requires tripping the reactor and tripping the affected RCPs?

A. RCP lower seal temperature increases to 150 °F.

B. RCP RBCCW flow has been lost for greater than 5 minutes.

C. RCP lower bearing temperature increases to 175 °F.

D. RCP controlled bleedoff temperature increases to 170 °F.

<<

>>Question 4

K&A Rating: 015/017AK2.08 (2.6)

K&A Statement: APE 015/017 Reactor Coolant Pump (RCP) Malfunctions, AK2.08 Knowledge of the interrelationships between the Reactor Coolant Pump Malfunctions and the following: CCWS

Key Answer: **B**

Justification:

A. **Incorrect:** RCP lower seal temperature increases to 150 ˚F is not correct. The RCP lower seal temperature trip criteria is 170 ˚F.
Plausible: Examinee needs to understand the relative values for trip criteria of components.

B. **CORRECT:** Loss of RBCCW flow for greater than 5 minutes is correct.

C. **Incorrect:** RCP lower bearing temperature increases to 175 ºF is not correct. The lower bearing temperature trip criteria is 194 ˚F.
Plausible: Examinee needs to understand the relative values for trip criteria of components.

D. **Incorrect:** RCP controlled bleedoff temperature increases to 170 ºF is not correct. The RCP controlled bleedoff temperature trip criteria is 195 ˚F.
Plausible: Examinee needs to understand the relative values for trip criteria of components.

References: AOP 2564 rev. 005-00 Student Ref: None

Learning Objective: 283217 (05020-MB) Describe the actions required when the time restriction associated with a loss of RBCCW header is exceeded.

Question Source: Bank 0056182-MB (310917)

Question History: No record of use on an NRC exam. Made some minor changes to enhance question and wrote justifications.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

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

Comments (Question 4): 08/11/2020 jwr. Replaced question to address NRC feedback from their 7/24/20 review.

rewrote 5/21/2020 DF

<<

>>QUESTION 5

The plant is in MODE 5 after refueling. The RCS was drained to mid-loop to perform work. Current conditions are:

- RCS level is 3” above the Reactor Vessel Flange.
- Both S/Gs are at 70% level.

A loss of Shutdown Cooling occurs.

1. What is the Inventory status of the RCS? AND

2. What is the method of RCS Heat Removal?

	1.	2.
A.	Decreased Inventory.	Reflux Boiling.
B.	Decreased Inventory.	Natural Circulation.
C.	Reduced Inventory Operations.	Natural Circulation.
D.	Reduced Inventory Operations.	Reflux Boiling.

<<

>>QUESTION 5

K&A Rating: AK1.01 (RO 3.9, SRO 4.3)

K&A Statement: APE 025, Loss of Residual Heat Removal System (RHRS), Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation.

Key Answer: **A**

Justification: The plant is in Decreased Inventory (greater than 3" below the RV flange). Since mid-loop work was performed, the S/G U-tubes have been drained. On a loss of SDC, the RCS would be cooled by reflux boiling due to the empty U-tubes.

A. CORRECT: Decreased Inventory and reflux boiling

B. Incorrect: No natural circulation due to empty u-tubes
Plausible: might think natural circulation can develop

C. Incorrect: Not in Reduced Inventory Operation (RIO)
Plausible: might think in RIO (level lower than 3" below vessel flange)

D. Incorrect: Not in Reduced Inventory Operation (RIO)
Plausible: might think in RIO (level lower than 3" below vessel flange)

References: OU-M2-201, S/D Safety Assessment Checklist

Student Ref: NONE

Learning Objective: 283367 (MB-05838) Describe the Reflux Boiling Process

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41(b)(5) Facility operating characteristics during steady state and transient conditions…

Comments (Question 5): 08/15/20 jwr. Angelo's validation on 8/14/20 identified that the correct answer was not correct because there wasn't a bubble in the Pressurizer. The correct answer is decreased inventory and reflux boiling. Answer order was changed to maintain "A" correct and the stem was changed to clearly indicate SG tubes were drained.

Formatted answers to common format (numbers at top of columns per NRC comments DF 7/30/20
<<

>>QUESTION 6

The plant is operating at 100% power. NI power starts lowering.

Diagnose the cause for lowering power.

A. ‘A’ Boric Acid Pump, P19A, is started.

B. BA MAKEUP VLV, CH-210Y failed open.

C. LTDN HX TCV RB-402 failed open.

D. LTDN HX TCV RB-402 failed closed.

>>QUESTION 6

K&A Rating: APE 026, AA1.02 (RO 3.2, SRO 3.3)

K&A Statement: APE 026, Loss of Component Cooling System (CCW). AA1 Ability to operate and/or monitor the following as they apply to the Loss of Component Cooling Water (CCW): AA1.02 Loads on the CCWS in the control room

Key Answer: **D**

Justification:

A. **Incorrect:** Starting the Boric Acid Pump won't inject boron without CH-514 or CH-210Y opened
Plausible: might think the BA pumps will push water regardless of valve position

B. **Incorrect:** CH-210Y failing open won't inject boron w/o BA pumps running
Plausible: might think pressure ΔP would cause BA to flow

C. **Incorrect**: RB-402 opening will remove boron from L/D
Plausible: might think temperature increase removes boron

D. **CORRECT:** RB-402 closing causes temperature increase, higher temperature will strip b boron from IX causing BA concentration in RCS to increase → power lowers

References: Lesson Plan CVC-00-C, Chemical and Volume Control System

Student Ref: NONE

Learning Objective: 281167 (MB-02358) Describe the effects on the CVCS system including a loss or malfunction of the following: B. RBCCW

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR55.41(b)(5), Facility operating characteristics, including…reactivity changes…

Comments (Question 6):02SEP20 DF – Changed A to more plausible answer based on Don Jackson's review
<<

>>QUESTION 7

The plant is operating at 100% power.

- Pressurize pressure is at 2250 psia.
- The selected Pressurizer Pressure Controller setpoint is 2250 psia.
- The selected Pressurizer Pressure Controller output is 30%.

Describe the effect on the plant if the controller output were to fail to 50%.

Spray Valves will be ___(1)___.
Proportional heaters will be at ___(2)___.
Pressurizer pressure will stabilize ___(3)___.

	(1)	(2)	(3)
A	partially open	minimum output	below setpoint
B	partially open	minimum output	at setpoint
C	closed	maximum output	below setpoint
D	closed	maximum output	at setpoint

<<

>>QUESTION 7

K&A Rating: 027AK2.03 (2.6)

K&A Statement: Pressurizer Pressure Control System Malfunction. Knowledge of the interrelationships between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners

Key Answer: **A**

Justification:

A. **CORRECT**: The proportional heaters will go to minimum and spray valves will open slightly. The pressure will drop and cause Backup Heaters to energize at 2200 psia. Because the Backup Heaters do not turn off till 2225 psia, pressure will stabilize slightly above 2200 psia (below setpoint) with the Backup Heaters energized. This is what is done to Force Pressurizer Sprays, i.e. the controlling Pressurizer Pressure Controller setpoint is lowered until the Controller output is between 40 and 50% output and the Spray Valve Controllers output signal starts to rise. This is equivalent to lowering the Pressurizer Pressure Controller output around 50 psia.

B. **Incorrect:** Pressurizer pressure will lower and will stabilize a little above 2200 psia. It will not stabilize at setpoint.
Plausible: Response of the Spray valves and Proportional Heaters is correct. The candidate could reasonably believe Pressurizer pressure will restore to 2250 psia based on Backup heaters being on since this is essentially Forcing Pressurizer Sprays. But the different is when Forcing Sprays the Backup Heaters are manually turned on. In this case the Backup Heaters are automatically coming on at 2200 psia and shutting off at 2225 psia.

C. **Incorrect**: Spray valves will not be closed and Proportional Heaters will not be at maximum output. The controller output failing to 50% will partially open the Spray valves and drive Proportional Heater output to zero.
Plausible: The candidate could reason that as pressure lowers the Spray Valves will close and the Proportional Heaters will energize, and this will raise pressure back to slightly below setpoint since it is a proportional controller. And the Proportional Heaters and Spray valves control off deviation from setpoint.

D. **Incorrect:** Pressurizer pressure will not restore to setpoint (~2250 psia). The setpoint was essentially lowered when the controller output failed to 50%.
Plausible: The candidate may not understand that the higher the Controller output lowers Proportional Heater output and raises the signal to open the Spray Valves. It is reasonable that it would operate such that higher Controller output would raise the output to the Proportional Heaters and lower the output to the Spray Valves.

References: OP 2204 Attachment 3 & 10 Rev 043, PLC-01-C Rev 4 Chg 6,
Student Ref: NONE

Learning Objective: 281915 (02996-MB) Given a change in a pressurizer level or pressure controller output, describe how the controlled components respond to that change and the effect that component response will have on the CVCS and the RCS.

Question Source: Bank 413193 (0085558-MB)

Question History: No history of use on an NRC exam and the stem and all the answers were modified.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7 / 45.7

Comments (Question 7): 08/19/2020 jwr. Post NRC validation identified that this question needed a time frame (if Backup Heaters are used since they would not come on until pressure lowered to 2200 psia) included to make it clearer or replace Backup Heaters with Spray Valves (since they would immediately go partially open). The question was changed to replace Backup Heaters with Spray valves.

6/21/20 jwr – John W suggested replacing the backup heaters with a spray valve. Left as is. 6/8/20 jwr – put initial setpoint in stem. Browning comment. 5/31/20 jwr. PF comment that four pieces for each answer is too much. Pete recommended removing column one since it is not needed and changing pressures in last column to at setpoint or below setpoint would be better. Changes made per comments.
<<

>> QUESTION 8

The plant was operating at 50% power when a Reactor Coolant Pump trips.

The CEAs do NOT insert.

What indicates that an ATWS occurred and actions taken in EOP 2525, Standard Post Trip Actions, were not successful?

A. Main Turbine Control Valves remain open.

B. Trip Circuit Breakers 1, 2, 5, and 6 remain closed.

C. NI Power indicates >15% and stable (SUR = 0 dpm).

D. Only 2 CEAs do not fully insert.

<<

>>QUESTION 8

K&A Rating: EA2 (RO 4.4)

K&A Statement: EPE 029EA2.01, Anticipated Transient Without Scram (ATWS). Ability to determine or interpret the following as they apply to a ATWS: EA2 Reactor nuclear instrumentation

Key Answer: **C**

Justification: A reactor trip occurs when electrical power is removed from the CEA drive mechanisms causing the CEAs to drop into the core. An Anticipated Transient Without Trip occurs when the CEDMs don't lose power. In EOP 2525, the operator will first attempt to open to open the TCBs, then open the MG set output contactors if the CEAs have still not inserted. The last resort is to send an operator to locally open the TCBs

A. **Incorrect:** This is not an indication that the reactor did not trip**.**
Plausible: The Main Control valves close on a plant trip.

B. **Incorrect:** The TCBs open to trip the reactor. On an ATWS, the TCBs may not open.
Plausible: Might not think of the step that opens the MG set output contactors

C. **CORRECT:** A reactor trip causes nuclear power (due to fission) to lower < 1% nearly instantaneously. Even when when considering decay heat, 15% power would indicate the CEAs did not insert into the core.

D. **Incorrect**: The reactor is considered tripped even if 2 CEA don't fully insert.
Plausible: The candidate may think 2 CEAs stuck out constitutes a failure of the reactor to trip due to the requirement to emergency borate if 2 CEAs don't full insert.

References: EOP 2525 Standard Post Trip Actions
Student Ref: NONE

Learning Objective: 283650 (MB-05425) Given a set of plant conditions, determine the actions required in accordance with the Immediate Actions of EOP 2525, Standard Post Trip Actions
Question Source: New
Question History: New
Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR55.41(b)(6) Design, components, and functions of reactivity control mechanisms and instrumentation

Comments (Question 8):
19AUG20 DF 100% miss rate since last edit, rewrote stem to make more operator-centric (actions taken in EOP 2525), changed B to state all TCBs remain closed. Added (SUR = 0 dpm) after discussion with D. Silk.
6/9/20 changed C. from 5% to 15%. Edited justification based on NRC comments

<<

>>QUESTION 9

The plant is being depressurized in accordance with EOP 2534, Steam Generator Tube Rupture.

- RCS T_{COLD} = 420 ºF
- RCS pressure = 500 psia
- Affected S/G pressure = 460 psia

Given:

EOP 2541, Appendix 2, Figure 1, RCS P/T Curve
EOP 2541, Appendix 2, Figure 2, RCP NPSH Curve

What actions should be taken?

A. Continue depressurizing the plant, RCS pressure is too high.

B. Stop depressurizing the RCS, RCS pressure is too low.

C. Increase the cooldown rate, RCS pressure is too high.

D. Secure any running RCPs, RCS pressure is too low.

<<

>>QUESTION 9

K&A Rating: EPE 038, G2.1.25 (RO 3.9)

K&A Statement: EPE 039 Steam Generator Tube Rupture, G2.1 Conduct of Operations, 2.1.25 Ability to interpret reference materials such as graphs, curves, tables, etc.

Key Answer: **D**

Justification:

A. **Incorrect:** RCS pressure meets depressurization requirements
Plausible: Might think there's too much margin for P/T curve

B. **Incorrect:** RCS pressure meets depressurization requirements
Plausible: reasonable to think stopping depressurization is plausible

C. **Incorrect**: RCS pressure meets depressurization requirements
Plausible: Might think there's too much margin for P/T curve

D. **CORRECT:** Pressure doesn't meet RCP NPSH requirements

References: EOP 2534, step 10, Reduce and Control RCS Pressure
EOP 2541, Appendix 2, Figure 1, RCS P/T Curve
EOP 2541, Appendix 2, Figure 2, RCP NPSH Curve

Student Ref: EOP 2541, Appendix 2, Figure 1, RCS P/T Curve
EOP 2541, Appendix 2, Figure 2, RCP NPSH Curve

Learning Objective: 283824 (MB-05780) Outline and explain the bass for the major action steps for EOP 2534, Steam Generator Tube Rupture

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR

Comments (Question 9):
<<

>>QUESTION 10

A Main Steam Line Break (MSLB) has occurred on the No. 2 Steam Generator (SG) inside Containment.

The _____(1)_____ is closed by _____(2)_____ to maintain SG No. 1 available as a heat sink in the event the No. 1 SG Main Steam Isolation Valve (MSIV) does not close.

A. (1) SG #1 Non Return Check Valve
   (2) Reverse flow

B. (1) SG #2 Non Return Check Valve
   (2) Reverse flow

C. (1) SG #1 Non Return Check Valve
   (2) Main Steam Isolation (MSI) signal

D. (1) SG #2 Non Return Check Valve
   (2) Main Steam Isolation (MSI) signal

<<

>>QUESTION 10

K&A Rating: 040AK3.03 (3.2)

K&A Statement: Steam Line Rupture. Knowledge of the reasons for the following responses as they apply to the Steam Line Rupture: Steam line non-return valves

Key Answer: **B**

Justification:

A. **Incorrect:** Flow from the good SG (No. 1) would push open its non-return check valve (MS-1A) and if the faulted SG's MSIV (MS-64B) did not close the good SG (No. 1) would blow down through the faulted SG (No. 2).
Plausible: Examinee could think that the valve(s) associated with its SG protect that SG. That SG No. 1 is protected by MS-64A and MS-1A. And that air pressure holds the non-return check valves up out of flow like it does with the MSIVs.

B. **CORRECT:** Reverse flow (flow from the good SG) will close the non-return check valve (MS-1B) on the faulted SG (No. 2) to maintain the good SG (No. 1) as an RCS heat sink. The MSI signal only closes the MSIVs (MS-64A & B).

C. **Incorrect:** The non-return check valve does not receive a MSI signal.
Plausible: Examinee could think that the valve(s) associated with its SG protect that SG. That SG No. 1 is protected by MS-64A and MS-1A. And that air pressure holds the non-return check valves up out of flow like it does with the MSIVs and that an MSI signal closes the valve.

D. **Incorrect**: The non-return check valves do not receive an MSI signal.
Plausible: The MSIVs and non-return checks are sometimes thought of as one valve since they are physically together so the the examinee good thing an MSI signal closes both valves.

References: MSS-00-C Rev 7 Chg 1, Student Ref: NONE

Learning Objective: 281720 (02883-MB) Describe the design feature(s) and/or the components of the Main Steam that prevent reverse flow of steam during a steam line break accident.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.5

Comments (Question 10): 08/06/2020. Made changes after NRC review on 7/24/20. The NRC had no comments but I discussed with the NRC that I wanted make changes to the answers to specify the SG #1 or SG #2 Non Return Check Valve instead of using the valve number. Also identified and changed that it is the No. 1 SG Main Steam Isolation Valve (MSIV) not the No. 2 SG that does not close in the stem.
6/21/20 jwr – changed this question again after John W got incorrect. Quest ion was a little confusing with the either or both and did not need to be. 6/9/20 jwr – Answer “C” had 64A which was incorrect for the #2 MSIV. Changed to 64B. 6/4/20 jwr - Question was changed to put it in a little simpler form. This was a comment for PF. The first OPS validators got it correct but both training validators got it incorrect.
<<

>>QUESTION 11

The plant is at 12% power with the Main Turbine at 1800 rpm.

All condensate is lost when the running pump trips.

The other Condensate pumps are not available.

How will the plant respond?

	Main Feed Pump Trips:	Reactor Trips on:	Turbine Trips on:
A.	Immediately	Low SG level	Reactor Trip
B.	Immediately	Turbine Trip	Low SG level
C.	After 30 seconds	Turbine Trip	Low SG level
D.	After 30 seconds	Low SG level	Reactor Trip

<<

>> QUESTION 11

K&A Rating: 054AK3.01 (4.1)

K&A Statement: Loss of Main Feedwater (MFW). Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Reactor and/or turbine trip, manual and automatic.

Key Answer: **D**

Justification:

A. Incorrect: The Main Feedwater pump (MFP) will not immediately trip. The low suction pressure trip has a 30 second time delay.
Plausible: Examinee must remember that there is a low suction pressure trip and that it has a 30 second time delay. The reactor will trip on low Steam Generator (SG) level and the reactor trip does cause a turbine.

B. Incorrect: The MFP will not immediately trip. The low suction pressure trip has a 30 second time delay. The reactor will not trip as a result of a turbine trip below 15% power. The turbine does not trip on low SG level.
Plausible: Examinee must remember that there is a low suction pressure trip and that it has a 30 second time delay. A turbine trip will cause a reactor trip at greater than or equal to 15% power. The turbine does trip on SG level but it is high level not low level.

C. Incorrect: The reactor will not trip as a result of a turbine trip below 15% power. The turbine does not trip on low SG level.
Plausible: A turbine trip will cause a reactor trip at greater than or equal to 15% power. The turbine does trip on SG level but it is high level not low level.

D. Correct: The MFP will trip on low suction pressure after 30 seconds (trip has a 30 second time delay). Steam Generator level will then lower to the automatic RPS trip setpoint of 49.5% and the reactor will trip. The turbine will then trip from the reactor trip under customer trip #1

References: ARP 2590D-013 Rev 002, ARP 2590C-062 Rev 000, ARP 2590C-055 Rev 000-02, MT-00-C Rev 7/1, ARP 2590C-033 Rev 000D, ARP 2590D-041 Rev 002
Student Ref: NONE

Learning Objective: 281651 (02675-MB) Predict the response of a SGFP for a low suction pressure condition as given in MFW-00-C.

Question Source: Bank

Question History: Ginna 2008 NRC exam question No. 10

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.10

Comments (Question 11):<<

>>QUESTION 12

The following plant conditions exist:

- A Station Blackout has occurred.
- Instrument Air System pressure is 0 psig.
- Bus 201A is deenergized.

Where MUST local manual control be established to maintain RCS Heat Removal?

A. Turbine Driven Auxiliary Feedwater Pump.

B. Condenser Dump Valves.

C. Atmospheric Dump Valves.

D. Auxiliary Feedwater Control Valves.

<<

>>QUESTION 12

K&A Rating: EPE 055, EK1.02 (RO 4.1)

K&A Statement: EPE 055, Loss of Offsite and Onsite Power (Station Blackout). EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling.

Key Answer: **C**

Justification:

A. **Incorrect:** DC power is used for control of the Turbine Drive Auxilairy Feedwater (TDAFW) pump. With Bus 201B energized control from the control room is maintained and local manual control is not required.
Plausible: The TDAFW pump is controlled from the control room using DC power. Bus 201A is one of the sources of DC power that can be used to control the TDAFW pump.

B. **Incorrect:** The Condenser Dump Valves are not available for RCS Heat Removal since the MSIVs will be closed on both loss of Instrument Air and Bus 201A being de-energized
Plausible: Might think Condenser is still available

C. **CORRECT:** Atmospheric Dump Valves (ADVs) will fail shut as a result of the lost of Instrument Air (IA). Local manual control will have to be taken to open the ADVs to restore the Steam Generators as a heat sink and establish Natural Circulation cooling of the reactor core.

D. **Incorrect**: The Auxiliary Feedwater Control (AFWC) valves will fail open on a lost of IA and the #1 Steam Generator (AFWC) valve will fail open as a result of Bus 201A being lost. But local manual control is not required because the TDAFW pump speed can be controlled from the control room by lowering or raising TDAFW pump speed. Also the (AFWC) valves have backup air bottles to allow operation from the control room if IA is lost.
Plausible: The Auxiliary Feedwater Control (AFWC) valves will fail open on a lost of IA.

References: EOP 2530, Station Blackout rev. 017-00, EOP 2541, Appendix 7 TDAFW Pump Abnormal Startup rev. 001-00, AOP 2563 Loss of Instrument Air rev. 010-01, AOP 2505A Loss of Vital 125 VDC Bus 201A rev. 003.

Student Ref: None

Learning Objective: 283760 (MB-05912) Outline and explain the bases for the major actions in EOP 2530, Station Blackout.

Question Source: Bank 0156296-MB (451418)

Question History: NRC 2000 NRC exam Question #52, 055EK1.02, Comprehensive

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

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

Comments (Question 12):

02SEP20 Changed B from L/D flow control valves to Condenser steam dumps based on Don Jackson comments (L/D not plausible for RCS Heat Removal)

08/04/2020 jwr question replacement. The K/A and question were replaced in response to the feedback provided by the NRC on 07/24/2020. Initial question submitted was low level and an appropriate level question good not be written. Bank question stem was changed slightly to “maintain RCS heat removal” instead of “maintain Safety Functions”. Changed answer “B” to Letdown Isolation valves and provided a value for IA pressure in stem. Also added detail to justification.

<<

>>QUESTION 13

The plant is operating at 100% power.

The plant experiences a Loss of Offsite Power (LOOP).

How does 2-SW-8.1A, ‘A’ RBCCW HX SW TCV, respond to the event?

A. Valve is approximately the same position maintaining ‘A’ RBCCW header temperature.

B. Valve is modulating closed due to a lower heat load caused by the LOOP.

C. Valve is modulating open due to higher heat load caused by the LOOP.

D. Valve is failed open due to LNP signal.

<<

>>QUESTION 13

K&A Rating: APE 056, AA1.29 (RO 2.7)

K&A Statement: APE 056, Loss of Offsite Power AA1.29, Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power: CCW het exchanger temperature control valves.

Key Answer: **B**

Justification:

A. **Incorrect:** The TCV will be throttling closed due to RCPs tripping on LOOP
Plausible: Might think plant trip causes less load on RBCCW
B. **Correct:** RBCCW heat load lowers due to RCPs tripping

C. **Incorrect**: RBCCW heat load lowers during to a LOOP
Plausible: might think natural circ will cause RBCCW heat load to increase
D. **Incorrect:** no SIAS/LNP signal
Plausible: might think get SIAS/LNP on all ESF modules

References: 25203-26008, Service Water P&ID Run on simulator 5/9/20

Student Ref: NONE

Learning Objective: 281690 (MB-02710) Describe the effects of a loss or malfunction of the following on the operation of the Turbine Generator:
C) 125 VDC control power

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: 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 (Q13): 09/02/2020 dmf. Editorial fix identified by Don Jackson. Stem had "occurrs" instead of "occurs".

18AUG20 DF Removed ‘SIAS’ from D (used to read … SIAS/LNP...)

Changed correct answer to b. after running on simulator DF 5/11/20

<<

>>QUESTION 14

The plant is operating at 100% power.

A loss of VA-20 occurs.

The plant trips due to a complete loss of condenser vacuum.

How does the Loss of VA-20 affect the implementation of EOP 2525?

A. #2 S/G pressure is manually controlled locally.

B. Facility 1 Control Room A/C needs to be started.

C. Emergency Boration can only be accomplished using the Gravity Feed valves.

D. RCS temperature control is maintained by PIC-4216.

<<

>>QUESTION 14

K&A Rating: APE 057, AK3.01 (RO 4.1)

K&A Statement: APE 057, Loss of Vital AC Electrical Instrument Bus, AK3.01, Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac instrument bus

Key Answer: **A**

Justification:

A. **CORRECT:** VA-20 powers PT-4224, loss of controller operations on C-05, C-21 and C-10

B. **Incorrect:** Facility 2 dampers reposition to recirc position
Plausible: might think CRAC needs to be swapped to Facility 1

C. **Incorrect**: Gravity Feed valves are powered from Bus 22E (Facility 1)
Plausible: might think valves receive a signal from Facility 2 power source.

D. **Incorrect:** On a loss of vacuum, the steam dump control circuit is disabled .
Plausible: might think the steam dumps still can operate with no vacuum.

References: AOP 2504D, Loss of 120 VAC Vital Instrument Panel VA-20

Student Ref: NONE

Learning Objective: 282852 (MB-05738) Given a set of conditions during a loss of Non-Vital Instrument Panels VR-11 & VR-21 and Vital Instrument Panels VA-10, VA-20, VA-30, and VA-40, determine equipment limitations caused by these conditions

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: 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 (Question 14):

19AUG20 DF Added information to stem that vacuum is lost to give a reason for entering 2525 that closes the MSIVs

Updated in response to NRC comments. DF 11AUG20

<<

>>QUESTION 15

A plant Cooldown is in progress in accordance with OP 2207, "Plant Cooldown".

- ‘A’ Train of SDC is in operation.
- The SDC valves are being controlled from C-01.
- A loss of Vital 125VDC Instrument Panel DV10 has occurred.

Which one of the following actions will occur as a result of the loss of DV10?

A. “A” RBCCW HX RBCCW outlet temperature will rise.

B. “A” SDC HX RBCCW flow will lower.

C. SDC return temperature will rise.

D. SDC system flow will lower.

<<

>>QUESTION 15

K&A Rating: 058AA1.03 (3.1)

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

Key Answer: **C**

Justification:

A. **Incorrect:** 2-SW-8.1A, "A" RBCCW HX Temperature Control Valve (TV6308), fails OPEN. "A" RBCCW HX RBCCW outlet temperature will lower. SDC temperature control valve, SI-657 also fails closed which lowers the RBCCW heat load.
Plausible: Valve SW-8.1A fails open on loss of DV-10, not closed. And SDC temperature control valve, SI-657 fails closed which raises the SDC return temperature to the RCS.

B. **Incorrect:** "A" SDC HX RBCCW flow will lower will not lower. RBCCW flow through the SDC Hx is manually set by 2-RB-14A.
Plausible: The SDC temperature control valve, SI-657 fails closed which reduces the heat load to the RBCCW system.

C. **CORRECT**: 2-SI-657 SDC HX Flow Control Valve fails CLOSED. The closure of SI-657 stops SDC flow through the SDC Hx. SDC return temperature will rise due to lack of HX cooling.

D. **Incorrect:** 2-SI-306, SDC Total Flow Control Valve fails OPEN. Total SDC flow would rise, except that, per OP-2207, SDC is always placed in service with LPSI Loop Injection isolation valves SI-615, 625, 635, and 645 throttled. The throttled LPSI Loop Isolation valves ensure that a failure of SI-306 will not result in a high flow condition and a potential loss of the LPSI pump. Flow will either stay the same or rise very slightly. It will not lower.
Plausible: Valve 2-SI-306, fails on loss of DV-10 but open, not closed. Candidate could also confuse SI-657 and SI-306 failure positions, since SI-657 fails closed and SI-306 fails open.

References: AOP 2506A Rev 005, OP 2207 Rev 046-00 Student Ref: NONE

Learning Objective: 282908 (05727-MB) Given a set of plant conditions, explain how SDC responds to a loss of DV10 or DV20 per the load list attachment of AOP 2506A or B.

Question Source: Bank 453175 (0071660-MB)
Question History: NRC exam 2014 Q-15. Slight change to stem and enhanced justification.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7

Comments (Question 15): 08/6/2020 jwr. Made changes from NRC feedback 7/24/20. The NRC requested that instrument numbers be removed from answers and that the word “modification” be removed from the question history.

5/31/20 jwr - changed from operating in “automatic” to SDC valves are being “operated from C-01”. PF comment that the valves are not in automatic, they are in remote not manual. 5/18/20 jwr - added “Automatic”…i.e. SDC system is operating in Automatic… <<

>>QUESTION 16

The plant is operating at 100% power.

The crew enters AOP 2565, Loss of Service Water, to address a suspected Facility 1 Service Water system leak.

1. What would indicate that a Facility 1 Service Water leak exists? AND

2. Where is the location of the leak?

A. 1. Reactor Building Closed Cooling Water (RBCCW) Surge Tank level rising.
   2. "A" Reactor Building Closed Cooling Water Heat Exchanger.

B. 1. Turbine Building Closed Cooling Water (TBCCW) Surge Tank level rising.
   2. "A" Turbine Building Closed Cooling Water Heat Exchanger.

C. 1. VITAL AC SWGR ROOMS CLG COIL ISOL TROUBLE Main Board alarm.
   2. X-181A/B West 480V Switchgear Room Cooling Coil.

D. 1. VITAL AC SWGR ROOMS CLG COIL ISOL TROUBLE Main Board alarm.
   2. X-183 56' 4160V Switchgear Room Cooling Coil.

<<

>> QUESTION 16

K&A Rating: 062AA2.01 (2.9)

K&A Statement: Loss of Nuclear Service Water. Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: Location of a leak in the SWS.
Key Answer: **C**
Justification:

A. **Incorrect:** A leak in the "A" RBCCW Hx would result in RBCCW leaking into Service Water. Not Service Water leaking into RBCCW. RBCCW is at ~ 125 psig and Service Water at the Heat Exchangers is ~ 60 psig. Therefore not a SW leak.
Plausible: A rise in the RBCCW Surge tank does indicate a leak into RBCCW and the "A" RBCCW Hx is on Facility 1.

B. **Incorrect:** A leak in the "A" TBCCW Hx would result in TBCCW leaking into Service Water. Not Service Water leaking into TBCCW. TBCCW runs at ~ 84 psig (low pressure alarm is at 63.4 psig) and Service Water at the TB Heat Exchangers is ~ 45 psig (SW pump discharge pressure is ~ 45 psig from SP 2612A-003). Therefore not a SW leak. This is also not correct because the "B" Service Water header supplies the "A" TBCCW Hx.
Plausible: A rise in the TBCCW Surge tank does indicate a leak into TBCCW.

C. **CORRECT:** The VITAL AC SWGR ROOMS CLG COIL ISOL TROUBLE alarm on the Main board indicates a leak on one of (3) switchgear coolers. The alarm is due to water in the

cofferdam. The X-181A/B West 480V Switchgear Room Cooling Coil is normally aligned to Facility 1 Service Water since the room contains Facility 1 480 volt power (22A, B, & E).

D. **Incorrect**: X-183 56' 4160V Switchgear Room Cooling Coil is normally aligned to the Facility 2 Service Water header since it cools the Facility 2 4160V switchgear.
Plausible: The VITAL AC SWGR ROOMS CLG COIL ISOL TROUBLE alarm on the Main board would indicate a leak on X-183 56' 4160V Switchgear Room Cooling Coil.

References: AOP 2565 Rev 007-00, OP 2326A Rev 030, ARP 2590E-035 Rev 001, TBC-00-C Rev 5/2, RBC-00-C Rev 9/1, P&ID 25203-26008 SH 3 Rev 34, SP 2612A-005 Rev 005, OP 2326A-001 Rev 007 Student Ref: NONE

Learning Objective: 283253 (05045-MB) Outline the major actions for a loss of Service Water.
Question Source: New
Question History: New
Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X
10CFR55: CFR 43.5
Comments (Question 16): 08/06/2020 jwr. Completed changes recommended by the NRC on 7/24/20. The NRC recommended that the stem be changed to add "that" after "indicate'. I also added a question mark in the stem at the end of statement 1.
06/1/202 jwr. Validation comment from PF that the "A" TBCCW heat exchanger being feed from the "B" Service Water header was a little tricky and not needed for this question. Changed first part of answer "B" to make it also incorrect but plausible. 5/18/20 jwr - Incorporated NRC's comment on question structure, i.e. make stem and answers separated by (1) and (2) and capitalize AND in stem.<<

>>QUESTION 17

The plant is operating at 100% power.

What requires entry into EOP 2525, Standard Post Trip Actions?

A. Condenser backpressure rises to 5.5 “Hg absolute.

B. Instrument air pressure lowers to 75 psig.

C. Reactor power rises to 101.5% power.

D. An LNP occurs on Bus 24C, with an auto-start of the ‘A’ EDG.

<<

>>QUESTION 17

K&A Rating: APE 065 G2.4.1 (RO 4.6)

K&A Statement: APE 065 Loss of Instrument Air, G2.4.1 Knowledge of EOP entry conditions and immediate action steps

Key Answer: **B**

Justification:

A. **Incorrect:** Vacuum trip criteria is 7.5 “Hg
Plausible: 5 “Hg backpressure is trip criteria during if Loss of Vacuum actions are not successful

B. **CORRECT:** IA pressure <80 psig is Reactor Trip criteria

C. **Incorrect**: trip criteria is 105.8% power (admin limit 101.8% - 2 pre-trips)
Plausible: might think trip is required at this power level

D. **Incorrect:** trip not required for loss of vital 4160 bus
Plausible might think loss of bus necessitates reactor trip

References: AOP 2563, Loss of Instrument Air

Student Ref: NONE

Learning Objective: 283189 (MB-05702) State the conditions related to a loss instrument air which require tripping the reactor and explain the basis for this action

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility

Comments (Question 17):
<<

>>QUESTION 18

The plant is operating at 100% power.

ISO-NE notifies the unit that grid voltage is low and is forecast to go lower. ISO-NE requests the unit take actions to stabilize grid voltage.

In order to stabilize grid voltage, the BOP would place AUTO AC REG (CS-90) to the

___(1)___ position , causing MVARS to ___(2)___.

	(1)	(2)
A.	LOWER	rise
B.	LOWER	lower
C.	RAISE	rise
D.	RAISE	lower

<<

>>QUESTION 18

K&A Rating: APE077, AK1.03 (RO 3.3)

K&A Statement: APE 077 Generator Voltage and Electric Grid Disturbances, AK1.03 Knowledge of the following concepts as they apply to Generator Voltage and Electric Grid Disturbances: Over-excitation.

Key Answer: **C**

Justification: K/A is met as the question deals with actions taken to over-excite the generator field in response to grid disturbances.

A. **Incorrect:** CS-90 to LOWER under-excites the generator field and lowers MVARS
Plausible: might think you want to lower CS-90 because grid voltage is lowering

B. **Incorrect:** CS-90 to LOWER under-excites the generator field and lowers MVARS but VARS need to be raised in this situation.
Plausible: might think you want to lower CS-90 because grid voltage is lowering

C. **CORRECT:** Placing CS-90 to RAISE over-excites the generator field and increases generator MVARs

D. **Incorrect**: Placing CS-90 to LOWER over-excites the generator field and raises MVARS
Plausible: might think raising field voltage raises MW and lowers MVARs

References: OP 2204, Load Changes, Att. 16
Student Ref: NONE

Learning Objective: 281682 (MB-02691) Given that the Turbine Generator is synchronized to the grid, describe the reaction of the generator to:
a) a change in the Main Turbine load demand, or
b) a change in AC voltage regulator setpoint

Question Source: DC Cook 2012 NRC SRO Written Exam (#58)

Question History: Modified

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 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 (Question 18): removed column 2 (generator excitation) and added K/A justification in response to NRC comments DF 7/30/20
<<

>>QUESTION 19

The plant is at 70% raising power. Group 7 Control Element Assemblies (CEAs) are at 154 steps.

As the Group 7 CEAs are being withdrawn the center CEA stops moving.

What does the CEA Malfunction AOP check that would indicate the center CEA is misaligned from its group?

A. Incore Detector System.

B. Excore Nuclear Instruments.

C. RCS Cold Leg Temperature.

D. Axial Shape Index (ASI).

<<

>>QUESTION 19

K&A Rating: 005AA2.01 (3.3)

K&A Statement: Inoperable/Stuck Control Rod. Ability to determine and interpret the following as they apply to the Inoperable / Stuck Control Rod: Stuck or inoperable rod from in-core and ex-core NIS, in-core or loop temperature measurements.

Key Answer: **A**

Justification:

A. **CORRECT:** AOP 2556 CEA malfunctions, Attachment "D" step D.21 checks INCORE TILT and FRT. The Incore Detector system provides the best and most accurate indication of conditions in the core. It has (45) strings across the core with detectors at 20%, 40%, 60% and 80% of core height. The Incore Detector Monitoring system continuously provides a direct measure of the peaking factors and alarms for individual incore detector segments. The Incore Detector system would identify power in the area around the stuck CEA lower than other locations in that plane.

B. **Incorrect:** AOP 2556 CEA malfunctions, Attachment "D" step D.20 checks POWER RANGE Drawers which include the Excore detectors. But the center CEA not moving would not indicate anything abnormal on the Excores because the Excore quadrants would not deviate from each other. Excore Nuclear instrument indicate reactor power but would not be able to identify the center CEA was not moving out.
Plausible: Excore nuclear instruments detect power instantaneously and if the stuck CEA was on the periphery it would show a lower power on the closes detector and produce a deviation between the quadrants.

C. **Incorrect:** RCS Cold Leg temperature would not indicate whether a CEA was stuck and not aligned with its group and is not checked by the CEA Malfunction procedure to identify a CEA malfunction.
Plausible: On a dropped or slipped CEA RCS Cold Leg temperature will lower. The candidate may reason that a stuck CEA would have a similar response to a dropped or slipped CEA.

D. **Incorrect**: Axial Shape Index (ASI) is power (L-U/L+U). The change in ASI would be very minor and extremely hard to identify that a CEA was not moving.
Plausible: ASI would change because power in the upper half would be rising.

References: CPD-00-C Rev 6/2, BKG AOP 2556 Rev 022-00, AOP 2556 Rev 022, ARP 2590C-148 Rev 001, RRS-01-C Rev 5/3, Technical Specifications Rev through Chg 397, ARP 2590C-086 Rev 000-01, CPD-00-C Rev 6/2 Student Ref: NONE

Learning Objective: 281078 (MB-01419) Describe the methods available to monitor core power distribution parameters.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 43.5 /43.13

Comments (Question 19): 8/06/2020 jwr. The question was changed in response to the NRC's feedback on 7/24/20. The NRC requested that "best" be removed from the stem and link to a procedure be added. The stem was changed to remove "best" and reworded to include a procedure link.
6/21/20 jwr – Comment from John W that Tq (Tilt) would change if the center CEA was not moving out but that incore detectors are better indication. The stem was changed to ask what would be the best indication… Replaced answer "D" with Axial Shape Idex (ASI).
5/18/20 jwr - this was one of the (10) questions that were sent early to the NRC. The NRC's comment was that an examinee could throw out the two temperature answers because there are similar. Suggestion was to replace one of the temperature answers with TILT, FrT, or ASI as distracters. Replaced answer "D" with Azimuthal Power Tilt (Tq).
<<

>>QUESTION 20

The plant is at 50% power.

- The ‘C’ Charging Pump is operating.
- The Reactor Regulating System T_{AVE} calculated value fails to 568 °F.

How does Pressurizer Pressure (P_{PZR}) respond?

A. P_{PZR} increases (↑) due to charging flow greater than letdown flow.

B. P_{PZR} decreases (↓) due to charging flow less than letdown flow .

C. P_{PZR} increases (↑) due Backup heaters energizing.

D. P_{PZR} decreases (↓) due to spray valves opening.

>>QUESTION 20

K&A Rating: 028AK3.02 (2.9)

K&A Statement: Pressurizer (PZR) Level Control Malfunction. Knowledge of the reasons for the following responses as they apply to the Pressurizer Level Control Malfunctions: Relationships between PZR pressure increase and reactor makeup/letdown imbalance.

Key Answer: **A**

Justification: At 50% power, the T_{AVE} signal from Reactor Reg is 550°F which equates to a Pressurizer level setpoint of 53%. When the T_{AVE} signal fails to 568°F, that equates to 100% power pressurizer level setpoint of 65%. The PLCS senses level 12% less than setpoint and sends a start signal to the B/U charging pump and a close signal to the letdown flow control valves. The charging/letdown mismatch will cause P_{PZR} to increase.

A. **CORRECT:** Pressurizer pressure will increase due to a mismatch between charging and letdown {charging flow (88gpm) greater than letdown (28gpm) flow} and pressurizer level rising.

B. **Incorrect:** Pressurizer pressure will not lower
Plausible: may think level is greater than program and letdown flow is greater than charging flow causing pressure to decrease

C. **Incorrect:**..The Backup heaters will not energize
Plausible: Might think level is level is greater than program and B/U heaters energized on insurge.

D. **Incorrect**: The spray valves won't open
Plausible: may confuse pressure control with level control and think the spray valves open due to the high signal

References: OP 2204, Load Changes
Attachment 4 Pressurizer Level Control Program
Attachment 5, Pressrizer Level Setpoint Program
Student Ref: NONE

Learning Objective: 281907 (02982-MB) Given the plant with a steam bubble in the pressurizer, and given a pressurizer level deviation from setpoint, describe the response of the Pressurizer Level Control System including setpoints (± 0.2%).

Question Source: New
Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41(b)(5) Facility operating characteristics during steady state and transient conditions…

Comments (Question 20):

11AUG20 DF. Completed changes in response to the NRC's feedback on 7/24/2020. The NRC commented that this question is not straight forward. That you should not have to read the answers to understand what the question is asking. A new question was written to address the NRC's comment. They also commented that we should review our bank. I reviewed our bank and the industry bank and there are no questions for this K/A in either bank.

5/19/20 jwr - this was one of the (10) questions that were sent early to the NRC. The NRC comment was that it was hard to understand what the question was asking. The initial OPS validation confirmed this. The NRC's suggestion was to ask something like what letdown system response and malfunction will result in a Pressurizer pressure change…, and put the answers in three columns. Again the OPS initial validation provided the same feedback. Wrote entirely new question based on the NRC and OPS feedback. 6/22 changed stem to read differently.

<<

>>QUESTION 21

How does AOP 2577, “Fuel Handling Accident”, direct ventilation be aligned in response to a fuel handling accident, when containment purge is in service?

A. Push CPVIS actuation pushbuttons (C01), and
Place the Control Room Air Condition System in “RECIRC”.

B. Push CPVIS actuation pushbuttons (C01), and
Place the Enclosure Building Filtration system on the Enclosure Building.

C. Close the Containment Purge Dampers 2-AC-5, 6, 7 & 8, and
Place the Control Room Air Condition System in “RECIRC”.

D. Close the containment purge dampers 2-AC-5, 6, 7 & 8, and
Place the Enclosure Building Filtration system on the Enclosure Building.

<<

>> QUESTION 21

K&A Rating: 036AA1.01 (3.3)

K&A Statement: Fuel Handling Accident. Ability to operate and / or monitor the following as they apply to the Fuel Handling Incidents: Reactor building containment purge ventilation system.

Key Answer: **C**

Justification:

A. Incorrect: Pushing the CPVIS actuation pushbuttons on C01 is not correct. There are no manual actuation pushbutton for CPVIS on C01.
Plausible: The CPVIS actuation signal will close the containment purge dampers 2-AC-5, 6, 7, & 8. It is one of the few signals that doesn't have an actuation pushbutton. Placing Control Room Air Conditioning (CRAC) in RECIRC is correct.

B. Incorrect: Pushing the CPVIS actuation pushbuttons on C01 is not correct. There are no manual actuation pushbuttons for CPVIS on C01. Placing the Enclosure Building Filtration system (EBFS) on the Enclosure Building is also not correct.
Plausible: The CPVIS actuation signal will close the containment purge dampers 2-AC-5, 6, 7, & 8. Placing the EBFS on the Enclosure Building makes sense because it would filter any leakage from containment with charcoal filters.

C. Correct: The fuel handling accident AOP specifies if purging of containment is in progress, verify the following are closed: 2-AC-5, 6, 7, & 8, containment outboard isolation dampers, and place at least one train of CRACS in the recirculation mode. There is no manual actuation pushbuttons for CPVIS on C01.

D. Incorrect: Placing the EBFS on the Enclosure Building is not correct.
Plausible: Placing the EBFS on the Enclosure Building makes sense because it would filter any leakage from containment with charcoal filters. Closing the containment purge dampers 2-AC-5, 6, 7, & 8 is correct.

References: AOP 2577 Rev 009, ESA-01-C Rev 4/1 Student Ref: NONE

Learning Objective: 283453 (05552-MB) Describe the general sequence of operations for AOP 2577, "Fuel Handling Accident"

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7

Comments (Question 21): 08/07/20 jwr. Added "when containment purge is in service" to the stem in response to the NRC feedback on 7/24/20.6/8/20 jwr - removed "to close Purge Dampers 2-AC-5, 6, 7, & 8" for answers "A" and "B". This was a comment for Browning. This was very close to the other answers. 5/8/20 jwr shortened stem<<

>>QUESTION 22

Technical Specifications limit primary to secondary LEAKAGE through any one steam generator to…

A. 1 GPM.

B. 10 GPM.

C. 75 GPD.

D. 125 GPD.

>>QUESTION 22

K&A Rating: APE 37, G2.2.38 (RO 3.6)

K&A Statement: APE 037 Steam Generator Tube Leak, G2.2.38 Knowledge of conditions and limitations in the facility license

Key Answer: **C**

Justification:

A. **Incorrect:** This is the Tech Spec limit for IDENTIFIED LEAKAGE
Plausible: may confuse this limit with limit for primary to secondary LEAKAGE

B. **Incorrect:** This is the Tech Spec limit for IDENTIFIED LEAKAGE
Plausible: may confuse this limit with limit for primary to secondary LEAKAGE

C. **CORRECT:** Tech Spec 3.4.6.2 limits primary to secondary leakage through any one S/G to 75 GPD

D. **Incorrect**: 150 GPD would be the limit for both S/Gs
Plausible: The student may think 125 GPD would be acceptable for leakage through 2 S/Gs.

References: Unit 2 Technical Specifications 3.4.6.2

Student Ref: NONE

Learning Objective: 283309 (MB-05775) Given a set of plant conditions, determine Technical Specification applicability concerning AOP 2569, Steam Generator Tube Leak.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility

Comments (Question 22): Changed answer from 150 gph to 150 gpd, swapped answer C & D, changed stem to include “through any one S/G” from comments made by NRC. DF 7/31/2020
<<

>>QUESTION 23

For the Area Radiation Monitors listed below, select the correct Setpoint, and Automatic Action.

	Radiation Monitor	Setpoint	Automatic Action
A.	CNTR RM VENT RAD, RM-9799A	1 mR/hr	Swaps CRACS to RECIRC Mode
B.	SFP AREA S.E., RM-8157	50 mR/hr	Initiates EBFS
C.	CTMT HI RAD (WEST), RM-8241	100 mR/hr	Isolates Hydrogen Purge Flowpath
D.	Personnel Access Area, RM-7890	2×10^5 cpm	Initiates Facility 2 EBFS

<<

>>QUESTION 23

K&A Rating: APE 061, AA1.01 (RO 3.6)

K&A Statement: APE 061 Area Radiation Monitoring (ARM) System Alarms, AA1.01 Ability to operate and/or monitor the following as they apply to the Area Radiation Monitoring (ARM) System Alarms: Automatic actuation

Key Answer: **A**

Justification
:

A. **CORRECT:** RM-9799A/B (CR Intake RM) swaps CRACs to RECIRC at 1mR/hr

B. **Incorrect:** Rad monitor initiates AEAS not EBFS.
Plausible: Might think EBFS is right because the ventilation systems are interconnected.

C. **Incorrect**: RM-8241 isolates Hydrogen Purge at 5 R/hr
Plausible: might think setpoint is lower than actual

D. **Incorrect:** RM-7890 has no automatic function
Plausible: Might think Enclosure Building RM can initiate EB Filtration Signal

References: ARP 2590A-159 CRACS in Auto Recirc Mode
Student Ref: NONE

Learning Objective:

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR

Comments (Question 23): 09/02/2020 jwr. Don Jackson's review recommended elimination of the second column. Question was left "as-is" since removal of this column would have created two correct answers.

6/21/20 jwr – changed SFP to 40 mr/hr to make question a little more straight forward. DF - modified due to commonality with Q64. DF 6/5/20. Modified Stem verbiage, swapped setpoint and Automatic Actions columns in response to NRC comments. DF 31JULY20
<<

>>QUESTION 24

When the fire suppression systems are actuated in the 45’ 6” or 25’ 6” cable spreading rooms,

1. What doors are required to be opened? AND
2. What equipment is being protected?

A. (1) East end of the 25’6 cable spreading room to stairway <u>and</u> access from bottom of stairway to outside.
   (2) Plant Process Computer power supply.

B. (1) East end of the 25’6 cable spreading room to stairway <u>and</u> access from bottom of stairway to outside.
   (2) “A” and “B” Batteries.

C. (1) East access to east DC SWGR room, access between DC SWGR rooms, <u>and</u> access from bottom of stairway to outside.
   (2) Plant Process Computer power supply.

D. (1) East access to east DC SWGR room, access between DC SWGR rooms, <u>and</u> access from bottom of stairway to outside.
   (2) “A” and “B” Batteries.

<<

>> QUESTION 24

K&A Rating: 067AA2.14 (3.2)

K&A Statement: Plant Fire On Site. Ability to determine and interpret the following as they apply to the Plant Fire on Site: Equipment that will be affected by fire suppression activities in each zone.

Key Answer: **B**

Justification:

A. Incorrect: The Plant Process Computer (PPC) power supply is not being protected by opening these doors. There are no openings from the cabling spreading room to the DC switchgear rooms that house the PPC power supply. The ceilings of the battery rooms have an opening to the cable spreading area and a door separates the battery room from the DC switchgear rooms.
Plausible: The cable spreading area is above the DC switchgear rooms that house the PPC power supply and you would want to maintain the PPC. Additionally opening the listed doors would drain water from these rooms.

B. Correct: The east end of the 25' 6" cable spreading room door to the stairway and the access from bottom of stairway to outside are opened to protect the "A" and "B" Battery Rooms. Opening these doors will allow drainage of accumulated water in the 25' 6" cable spreading room thus preventing overflow into the "A" and "B" Battery. Overflow may occur when levels exceed 3".

C. Incorrect: The Plant Process Computer (PPC) power supply is not what is being protected when the fire suppression system is activated in the 25' 6" or 45' 6" cable spreading room.
Plausible: The cable spreading area is above the DC switchgear rooms that house the PPC power supply and you would want to maintain the PPC. Additionally opening the listed doors would drain water from these rooms.

D. Incorrect: Opening the East Access to the East DC SWGR room, Access between the DC SWGR Rooms, and Access from the Stairway to Outside will not prevent water from getting into the battery rooms.
Plausible: Examinee could think battery rooms would drain through DC switchgear rooms.

References: AOP 2559 Rev 014, OP 2356 Rev 011-00, TRM Rev through change No. 180, PPC-00-C Rev 1/4, Drawing 25203-27032. Student Ref: NONE

Learning Objective: 283139 (05666-MB) State the action required for initiation of the fire suppression systems in the 45' or 25'6 cable spreading rooms.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 43.5

Comments (Question 24): 8/7/20 jwr. The stem was rewritten to address NRC comments on 7/24/20. The NRC's comment was that the stem was written with "backwards logic". The NRC provided a rewrite of the stem and it was used to change the stem.

6/4/20 jwr. The second validation with PF and HS identified that the vital instrument bus equipment was too close to the "A" and "B" batteries since the batteries are part of the vital instrument bus equipment. We discussed other equipment in the DC switchgear rooms and determined that the PPC would be a good plausible option to replace vital instrument bus equipment. Made changes discussed as well as editorial changes. 5/8/20 jwr added AC to make second part of answer for "A" and "C" "Vital AC Instrument Busses" instead of "Vital Instrument Busses". Clarification from OPS validation.<<

>>QUESTION 25

The Control Room is evacuated due to a fire. The Charging pumps are aligned as follows:

- "A" Charging pump is operating.
- "B" Charging is in standby as the 1st Backup, aligned to Facility 2.
- "C" Charging pump is in Pull-To-Lock.

How many Charging pumps can be controlled from the Fire Shutdown Panel (C10) at this time?

A. None.

B. One.

C. Two.

D. Three.

<<

>>QUESTION 25

K&A Rating: 068AK2.03 (2.9, 3.1)

K&A Statement: Control Room Evacuation. Knowledge of the interrelations between the Control Room Evacuation and the following: Controllers and positioners

Key Answer: **C**

Justification:

A. **Incorrect:** Both Facility Two pumps can be manually controlled from C10.
Plausible: The candidate could reason that Facility 1 equipment is available at C-10 but because the "B" Charging pump was not operating and "C" Charging pump was in PTL that neither are available.

B. **Incorrect:** Both Facility Two pumps can be manually controlled from C10.
Plausible: The candidate could reason that Facility 2 equipment is available at C-10 but that the "C" Charging pump is not available because it is in PTL.

C. **CORRECT**: Both Facility Two pumps can be manually controlled from C10. When control is shifted to C-10 all other switches in the circuit are bypassed. Therefore the "C" Charging pump is available, even when in PTL. And the "B" Charging pump is also available since it is electrically aligned to Facility 2.

D. **Incorrect:** Only the Facility Two pumps can be manually controlled from C10.
Plausible: The candidate could reason that Charging pumps are so important that all would be available to ensure a Charging pump would be available if one electrical facility was lost.

References: AOP 2579A Rev 013, A79-00-C Rev 3, CVC-00-C Rev 11 Ch 1
Student Ref: NONE

Learning Objective: 283010 (05676-MB) Predict how operator action or inaction affects plant and system conditions concerning a shutdown from outside the Control Room.

Question Source: 451492 (0253964-MB)

Question History: NRC 2001 exam Question #20, stem changed and plausibly statements enhanced.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7 / 45.7

Comments (Question 25):
<<

>>QUESTION 26

The plant is operating at 100% power.

AOP 2511, High Activity in RCS, is entered due to an increase in RCS fission product activity.

1. What action is required? AND

2. What is the reason for the action?

A. 1. Place a second Ion Exchanger (IX) in service.
   2. Limits the IX delta-P due to fission products.

B. 1. Downpower the unit to < 50% in one hour.
   2. Reduces the probability of additional clad failure.

C. 1. Start the two (2) non running Charging pumps and balance Letdown.
   2. Maximizes RCS purification.

D. 1. Contact the Reactor Engineer to obtain any power restrictions.
   2. Mitigates fuel damage.

<<

>>QUESTION 26

K&A Rating: 076AK3.05 (2.9)

K&A Statement: High Reactor Coolant Activity. Knowledge of the reasons for the following responses as they apply to the High Reactor Coolant Activity: Corrective actions as a result of high fission-product radioactivity level in the RCS

Key Answer: **D**

Justification:

A. **Incorrect:** The procedure does not direct placing an additional Ion Exchanger (IX) in service. The IX are maintained in series, therefore placing a second IX in service would not affect the delta-P across a single IX.
Plausible: Examinee may think that under the unusual situation of high activity in the RCS that an additional IX would need to be aligned due to increases fisson products.

B. **Incorrect:** The procedure does not require down powering the unit to < 50%.
Plausible: The Steam Generator Tube Leak procedure (AOP 2569) does require down powering the unit to < 50% in less than one hour to lower the probability of a Steam Generator tube failure. This is the correct response to a similar condition except one is increased activity in the RCS and one is increased activity in a Steam Generator.

C. **Incorrect**: Running all Charging pumps is not correct since operation of three Charging pumps is prohibited by procedure.
Plausible: The procedure directions starting additional Charging pumps and balancing Charging and Letdown flows. This is directed to maximize purification to lower RCS activity.

D. **CORRECT:** The procedure does direct contacting the Reactor Engineer if the RCS sample indicates an unexpected increase in fission products. The discussion section states that the Reactor Engineer may recommend power restrictions to mitigate fuel damage for elevated concentrations of fission products.

References: AOP 2511 Rev 000-04, OP 2304E Rev 024, A69-01-C Rev 3
Student Ref: NONE

Learning Objective: 282931 (06109-MB) Outline the major actions for Control Room personnel using AOP 2511, “High Activity in the RCS”.

Question Source: 8000012-MB

Question History: 452202(8000012-MB) NRC 2016 Q-25 and
451747 (8000012-MB) NRC 2008 Q-24
Significantly modified stem by changing pertinent condition in stem and all answers; effectively wrote entirely new question.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.5, 41.10 / 45.6 / 45.13

Comments (Question 26): 08/19/2020 jwr. Editorial change to answer “C”. Replaced “(2)” with “two”. Validator commented that he read it as the 2nd Charging pump.

08/07/2020 jwr. Completed the following change to address the NRC’s feedback on 7/24/20. Changed answer “C” to state “Start the (2) non running Charging pumps…” instead of “Start all (3) Charging pumps…”

6/4/20 jwr. Made changes as a result of comments from the second validation from PF and HS. Replaced answer “A” with placing a second Ion Exchanger in service. Their comment was that answer “A” to “Evacuate the Auxiliary Buidling…” is too close to the actual procedural requirement to “Notify Health Physics and Evacuate personnel from areas near letdown piping as applicable.”
5/18/20 jwr added (3) to answer “C”, OPS validation comment make it clear on how many pumps.
<<

>>QUESTION 27

The plant experienced a Loss of All Feedwater (LOAF)

- Once Through Cooling (OTC) has been established.
- EOP 2540D, Functional Recovery of Heat Removal, is in effect.
- HPSI Throttle criteria is being verified in order to reduce OTC flow.

What would prevent throttling HPSI flow?

A. ‘B’ S/G level = 160” and rising.

B. CET temperatures = 500ºF and stable.

C. Reactor vessel level = 29% and rising.

D. RCS subcooling = 35 ºF and stable.

<<

>>QUESTION 27

K&A Rating: E09, EK1.1 (RO 3.4)

K&A Statement: E09 Functional Recovery, EK1.1 Knowledge of the operational implications of the following concepts as they apply to (Functional Recovery): Components, capacity and function of emergency systems

Key Answer: **C**

Justification:

A. **Incorrect:** S/G available for heat removal and level being restored
Plausible: might think 165" required (165" number for implementing OTC)

B. **Incorrect:** CET temperature not a consideration for throttle/stop criteria
Plausible: May think 500°F is too hot

C. **CORRECT:** Reactor level is required to be **greater 43%** to throttle HPSI

D. **Incorrect:** Subcooling above minimum operating limit of RCS P/T curve.
Plausible: might think >35°F subcooling required.

References: EOP 2540D, Functional Recovery of Heat Removal (HR-3)

Student Ref: NONE

Learning Objective: 283935 (MB-05975) Describe the general approach used to recover lost safety functions:

a. EOP 2540D, Functional Recovery of Heat Removal

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility

Comments (Question 27): After Pete & Herb, changed to HPSI throttle/stop criteria from OTC termination criteria. Changed D to 30 degree subcooling 6/9/20

<<

>>QUESTION 28

Which of the following describes the INTERLOCKS that must be satisfied in order to start an RCP?

A. Minimum Lift pump oil pressure and minimum seal bleedoff flow.

B. Minimum Lift pump oil pressure and minimum RBCCW flow.

C. Minimum Pressurizer pressure and minimum seal bleedoff flow.

D. Minimum Pressurizer pressure and minimum RBCCW flow.

<<

>>QUESTION 28

K&A Rating: 003K4.04 (RO 2.8)

K&A Statement: 003 Reactor Coolant Pump System (RCPS), K4.04 Knowledge of RCPS design feature(s) and/or interlock(s) which provide for the following: Adequate cooling of RCP motors and seals

Key Answer: **B**
Justification:

A. **Incorrect:** There is no interlock associated with RCP seal controlled Bleedoff (CBO) flow. CBO flow is not an interlock required to be satisfied to start a RCP.
Plausible: CBO is an important parameter required to ensure adequate seal cooling. Abnormal CBO flow will cause alarms. And if CBO flow is not adequate procedures will direct that the RCP with the alarm be secured.

B. **CORRECT:** A RCP lift pump must be running with oil adequate oil pressure to the bearings and the RCP Cooling Water Flow Low alarm must be not lit (indicating adequate RBCCW flow) for the RCP to start when the breaker handswitch is operated.

C. **Incorrect**: There is no interlock associated with RCP seal controlled Bleedoff (CBO) flow and no interlock associated with minimum Pressurizer pressure.
Plausible: Some pumps have low suction pressure interlocks to protect the pump from being run when Net Positive Suction Pressure (NPSH) requirements are not met to prevent pump damage. And CBO is an important parameter required to ensure adequate seal cooling.

D. **Incorrect:** There is no interlock associated with minimum Pressurizer pressure.
Plausible: Some pumps have low suction pressure interlocks to protect the pump from being run when Net Positive Suction Pressure (NPSH) requirement are not met to prevent pump damage. Minimum RBCCW flow is correct.

References: RCS-00-C, Reactor Coolant System Lesson Plan
25203-32007 sh. Reactor Coolant Pump P40A
Student Ref: NONE
Learning Objective: 281984 (MB-03042) Describe the purpose and function of the interlock between the RCP and the oil lift pump.
Question Source: Bank 0054426-MB (not found in VISION Bank)
Question History: No record of it used on an NRC exam
Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:
10CFR55: 10CFR41.3 Mechanical components and design features of the reactor primary system
Comments (Question 28): 08/4/2020 jwr. Question and K/A replaced. NRC comment on their 07/24/2020 feedback was that plausibly of incorrect answers was low. 6/3 changed A from two seals to ARD. 6/22 changed B to read “any RCP seal …” from Vapor Seal

<<

>>QUESTION 29

The plant is at 100% power.

Vital Instrument Bus VA-10 is lost.

1. What source is borating the RCS? AND
2. What Immediate Operator Action is performed to mitigate the consequences?

A. (1) RWST.
   (2) Secure Letdown and Charging.

B (1) BAST.
   (2) Secure Letdown and Charging.

C. (1) RWST.
   (2) Lower main turbine load to stabilize T_{COLD}.

D. (1) BAST.
   (2) Lower main turbine load to stabilize T_{COLD}.

<<

>>QUESTION 29

K&A Rating: 004A2.10 (3.9)

K&A Statement: Chemical and Volume Control System. Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Inadvertent boration/dilution.

Key Answer: **A**

Justification:

A. Correct: The loss of VA-10 results in the loss of VCT level control LC-227. The loss of LC-227 swaps the Charging pump suction to the RWST and initiates boration to the RCS. The Immediate Operator actions (IOA) for a loss of VA-10 is to secure Letdown and Charging.

B. Incorrect: The loss of VA-10 does not initiate boration from the BAST to the RCS.
Plausible: Boration to the RCS would require securing Letdown and Charging. The examinee may know the loss of a vital instrument bus initiates boration but not from where.

C. Incorrect: The IOA for a loss of VA-10 does not include lowering main turbine load to stabilize T_c.
Plausible: Lowering main turbine load to stabilize T_c is done in the loss of VA-10 AOP and is an IOA for other events.

D. Incorrect: The loss of VA-10 does not initiate boration from the BAST to the RCS nor does the IOA for a loss of VA-10 include lowering main turbine load to stabilize T_c.
Plausible: Boration to the RCS would require securing Letdown and Charging. The examinee may know the loss of a vital instrument bus initiates boration but not from where. Lowering main turbine load to stabilize T_c is done in the loss of VA-10 AOP and is an IOA for other events.

References: BKG AOP 2504C Rev 009-00, BKG AOP 2504D Rev. 008-00, AOP 2585 Rev. 003-00
Student Ref: None

Learning Objective: 282855 (05741-MB) Outline the major actions for AOP 2504A/B/C/D/E/F, Loss of Non-Vital Instrument Panels VR-11 & VR-21 and Vital Instrument Panels VA-10, VA-20, VA-30, & VA-40.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.5

Comments (Question 29): 08/07/2020 jwr. The question was rewritten to address the NRC's 07/24/2020 comments. The stem and answers were changed. Their comment was that the stem has "backwards logic". They suggested the stem should be rewritten to say something like "VA-10 is lost what is the effect on inadvertent boration and what procedure action needs to be taken".

7/1/20 jwr. A major change to this question was made due to the high failure rate. The question was changed by removing having to choose between VA-10 and VA-20 to what results in boration from the RWST, thereby making it a better entry level RO question.

<<

>>QUESTION 30

The plant tripped from 100% power. EOP 2525, Standard Post Trip Actions, is in progress.

- Pressurizer level = 25% and stable.
- Pressurizer Pressure = 2200 psia and rising.
- Steam Generator levels lowered to 25% and are at 40% and rising.
- RCS T_C is 530 °F and slowly rising.

For the conditions above, state the action to be taken in accordance with EOP 2525, Standard Post Trip Actions.

A. Close the Atmospheric Dump Valves.

B. Take manual control of, and minimize Letdown.

C. Initiate Auto Auxiliary Feedwater.

D. Secure the Backup Heaters.

<<

>>QUESTION 30

K&A Rating: 004A2.35 (3.3)

K&A Statement: 004 Chemical and Volume Control System (CVCS), A2.35 Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and to (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Reactor trip

Key Answer: **B**

Justification:

A. Incorrect: Close the Atmospheric Dump valves (ADVs) closed is not correct. A T_C of 530 °F corresponds to 885 psia which is below the normal ADV setpoint of 920 psia. The RESPONSE NOT OBTAINED is taken if T_C is not between 530 – 535 °F.
Plausible: The examinee may think that the ADVs control temperature after the trip but with the Main Condenser available the "A" Condenser Dump valve controls temperature at ~ 532 °F, with a setpoint of 860 psig. With the temperature of 530 °F and rising the examinee may think that the Condenser Dump valves are not responding when in fact they are just opening further to respond to the increasing temperature.

B. CORRECT: Take manual control of, and minimize Letdown is correct answer. With Pressurizer level at 25% (15% below setpoint) Letdown should be at minimum. EOP 2525 specifies if Pressurizer level is not trending to 35 to 70% to take the RESPONSE NOT OBTAINED. Under this procedure step the automatic operation of the Pressurizer level control system is assessed and if it is not operating properly then manual control is directed to restore Pressurizer level. Letdown at maximum is an indication that the Pressurizer level control system is not operating properly.

C. Incorrect: Initiate Auto Auxiliary Feedwater Actuation is not correct. Override of Auto Auxiliary Feedwater Actuation is not a "RESPONSE NOT OBTAINED" action in EOP 2525 under Steam Generator level. It is a "RESPONSE NOT OBTAINED" under Steam Generator pressure if pressure is less than 572 psia, which it is not.
Plausible: There are steps to override an Auto Auxiliary Feedwater Actuation in 2525. And Followup Actions to EOP 2525 direct overriding an Auto Auxiliary Feedwater Actuation but it is an "ACTION/EXPECTED RESPONSE" not "RESPONSE NOT OBTAINED" and not done in EOP 2525.

D. Incorrect: Secure the Backup heaters is not correct. The Backup Heaters are on and are the expected response and an indication that the Pressurizer Heater Control system is functioning properly and therefore there is no need to take "RESPONSE NOT OBTAINED" actions. Backup Heaters automatically turn off when Pressurizer pressure increases to 2225 psia.
Plausible: Examinee could understand that the Backup Heaters automatically turn on at 2200 psia decreasing and think that they turn off at 2200 psia increasing. And if that was the case that the Backup Heaters are not functioning properly and "RESPONSE NOT OBTAINED" actions should be taken.

References: EOP 2525 rev. 028-00, OP 2204 rev. 045, EOP 2541, Appendix 4 rev. 008

Student Ref: None

Learning Objective: 283649 (05424-MB) Outline the Instruction and Contingency Actions for the Immediate Actions in EOP 2525, Standard Post Trip Actions.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.5 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 30):

02SEP20 DF Changed stem of question and answers to indicate what actions that would be taken with the stated conditions based on Don Jackson's review. DF. John updated the justification after Doug's changes on 9/9/20.

08/14/2020 jwr. Question replaced to address the NRC's 07/24/20 comments. The NRC's comment was do we need to provide Pressurizer Level. Upon further review by Doug and John no good plausible distracter could be developed. Due to this and to meet both parts of the K/A a new question was written.

6/9/20 changed stem to 525 ºF and stable.<<

>>QUESTION 31

The Plant is in Mode 6 with the “B” LPSI pump in service on Shutdown Cooling (SDC).

- The supply breaker to 24D opens.
- The diesel starts and energizes the bus.

1. What action, if any, is required to restore SDC to service? AND

2. Why?

A. 1. No action is necessary.
   2. The LPSI pump will start because it stays connected to the bus.

B. 1. No action is necessary.
   2. The LPSI pump will auto start as part of the LNP/SIAS response.

C. 1. The opposite train pump must be started.
   2. The B LPSI pump is locked out by the LNP signal.

D. 1. The “B” LPSI pump must be restarted.
   2. The auto start signal is not processed.

<<

>>QUESTION 31

K&A Rating: 005K2.01 (3.0)

K&A Statement: Residual Heat Removal (RHRS). Knowledge of bus power supplies to the following: RHR pumps
K/A match discussion. The examinee must understand where the power is being supplied from when in Mode 6 (RSST), how opening the supply breaker effects repowering the RHR pump, and how power is then supplied (B EDG) to the RHR pumps. They must understand that even when the bus has power that the RHR will not restart until operator action is taken.

Key Answer: **D**

Justification:

A. **Incorrect:** The LPSI pump will not restart.
Plausible: The candidate could reason that since the LPSI pump was running when power was lost that it simply restarts when power was restored. Lots of equipment responds in this way.

B. **Incorrect:** The LPSI pump will not auto start.
Plausible: The candidate could reason that since the EDG automatically energized the bus the LPSI pump will automatically start through the ESAS LNP/SIAS logic.

C. **Incorrect**: The running LPSI pump can be restarted once power is restored.
Plausible: The candidate could reason that part of the LNP response is to load shed the de-energized bus and reason that it stays locked out since there is not an active SIAS signal to start it.

D. **CORRECT:** When either 24C or 24D is lost, while in SDC, the bus will automatically re-energized but the LPSI pump will not automatically restart. This is due to the SIAS block and because the ESAS has not generated and auto start signal (i.e. SIAS). Therefore the operator must manually restart the associated LPSI pump.

References: SDC-00-C Rev 6 /1 Student Ref: NONE

Learning Objective: 282215 (03179-MB) Describe the effects on the Shutdown Cooling System of a loss or malfunction of the following: A) 4.16 KVAC Electrical Distribution System.

Question Source: Bank

Question History: 315898, 413516, (0078189-MB) modified. No record of being on an NRC exam. Revised question with minor modification.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7

Comments (Question 31): 08/11/2020 jwr. Changed part 2 of the stem from “What is the reason for this action” to “Why”. This was recommended by the NRC on the 8/11/20 question review call.

8/7/20 jwr. Completed a minor change to the question stem and added a discussion under the K/A to address the NRCs feedback provided on 7/24/20. The NRC had questioned the K/A match, but after discussion they agreed that the question matched the K/A. They requested that an explanation be added to how the question meets the K/A under the K/A in the justification.
6/10/20 df - Changed D2 to an “auto start signal is not processed”. Comment from John W.
<<

>>QUESTION 32

The plant is in MODE 5 preparing for entry into MODE 4. SDC is in service with Concurrent RCP Operations.

A 300 gpm tube leak develops in X23A, ‘A’ SDC Heat Exchanger.

Describe the effect on the following:

1. T351Y, SDC TO RCS, (C-01), temperature will _____.
2. FI-6043, SDC HX A RBCCW OUT (C-06), flow will _____.

	T351Y	FI-6043
A	rise	lower
B	lower	lower
C	rise	remain constant
D	lower	remain constant

<<

>>QUESTION 32

K&A Rating: 005 K6.03 (RO 2.5)

K&A Statement: 005 Residual heat Removal System (RHRS), K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger.

Key Answer: **C**

Justification: A tube leak in a SDC Hx during Concurrent operations causes RCS water (~300 psia) to flow into the RBCCW system (~120 psig). The RCS is at a higher temperature (~200F) than RBCCW (~75F). The RCS flowing into the SDC Hx will restrict the RBCCW flow through the Hx, reducing the heat transfer that occurs within the heat exchanger. Thus the RCS return (outlet of the Hx) temperature will rise. Even though RBCCW flow through the HX lowers due to the leak, indicated RBCCW flow remains constant since RBCCW is a closed system: the increase in pressure occurring inside the SDC Hx will be transmitted to both sides of the downstream throttle valve 2-RB-14A. This constant flow rate was validated on the simulator model.

A. **Incorrect:** RCS return temperature increases, FI-6043 will remain constant.
Plausible: May be thinking about actual RBCCW flow.

B. **Incorrect:** RCS return temperature increases
Plausible: May be thinking about actual RBCCW flow, may think RCS return temperature lowers due to lower RCS flow through Hx

C. **CORRECT:** See Justification above.

D. **Incorrect**: RCS return temperature increases, FI-6043 will remain constant.
Plausible: May think RCS return temperature lowers due to lower RCS flow through Hx

References: 25203-26015 sh.1 LPSI, Validated on glasstop simulato

Student Ref: NONE

Learning Objective: 283281 (MB-05505) Given a set of plant conditions indicative of an RCS leak, determine the most likely location of leakage.

Question Source: Braidwood NRC Exam 2011 - SRO

Question History: Modified

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility

Comments (Question 32): Swapped columns to put temperature first 6/9/20. Expanded justification for temperature lowering and constant flow based on NRC comments. DF 31JULY20

<<

>>QUESTION 33

The reactor was manually tripped due to an RCS leak.

- On the trip bus 22E was lost.
- After the loss of 22E, a SIAS was initiated.

What is the status of the ECCS injection valves?

A. (4) LPSI valves are open, (8) HPSI valves are open.

B. (2) LPSI valves are open, (8) HPSI valves are open.

C. (4) LPSI valves are open, (4) HPSI valves are open.

D. (2) LPSI valves are open, (4) HPSI valves are open.

<<

>>QUESTION 33

K&A Rating: 006 K2.04 (RO 3.6)

K&A Statement: 006 Emergency Core Cooling, K2.04 Knowledge of bus power supplies to the following: ESFAS-operated valves

Key Answer: **B**

Justification:

A. **Incorrect:** The (4) LPSI valves are not open. The Facility 1 LPSI injection valves 2-SI-615 and 2-SI-625 will not be open because the SIAS occurred after the lost of bus 22E. Therefore these MOVs did not have any power to open.
Plausible: Examine may not know the power supply to the LPSI injection valves. The second part is correct, all HPSI injection valves are open. Also may think there are (8) LPSI injection valves and half of them opened. Or there were problems during the 2020 refuel outage and one of the possible fixes (which wasn't done) was to leave all LPSI injection valves open like the HPSI injection valves.

B. **CORRECT**: The (2) Facility 2 LPSI injection valves open (since they have power) and all (8) HPSI injection valves are open (HPSI valves are maintained open) is correct.

C. **Incorrect**: The (4) LPSI valves are not open. The Facility 1 LPSI injection valves 2-SI-615 and 2-SI-625 will not be open because the SIAS occurred after the lost of bus 22E. Therefore these MOVs did not have any power to open. All (8) HPSI injection valves are open.
Plausible: Examine may not know the power supply to the LPSI injection valves. Also may think there are (8) LPSI injection valves and half of them opened. Or there were problems during the 2020 refuel outage and one of the possible fixes (which wasn't done) was to leave all LPSI injection valves open like the HPSI injection valves. May think either half the HPSI injection valves opened or that there are only (4) HPSI injection valves.

D. **Incorrect:** All (8) HPSI injection valves are not just (4).
Plausible Examinee might think only half the HPSI injection valves opened because of the lost of Facility 1 power similar to how the LPSI injection valves respond.

References: HPSI Lesson Text HPI-00-C and LPSI Lesson Text LPI-00-C.

Student Ref: NONE

Learning Objective: 281483 (02562-MB) and 281616 (02654-MB).

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 33): 6/30/20 jwr. Replaced with new question because it was 5 of 9 validators had missed. 6/22 change B from 638 to 648. Modified stem to one sentence
<<

>>QUESTION 34

The plant is in MODE 5, drawing a bubble.

Steam Generator U-tubes were NOT drained.

How are non-condensable gases removed from the RCS?

A. The pressurizer is vented continuously to the Primary Sample Sink.

B. The pressurizer is vented continuously to the Enclosure Building Purge System.

C. The Reactor Head vents are cycled to the Primary Sample Sink.

D. The Reactor Head vents are cycled to the Enclosure Building Purge System.

<<

>>QUESTION 34

K&A Rating: 007 K5.02 (RO 3.1)

K&A Statement: 007 Pressurizer Relief Tank/Quench Tank SYSTEM (PRTS), K5.02 Knowledge of the operational implications of the following of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR.

Key Answer: **A**

Justification:

A. **CORRECT:** The pressurizer is continuously vented to the primary sample sink while drawing a bubble

B. **Incorrect:** Piping can be aligned to vent the PZR to EBFS but isn't used for this evolution
Plausible: May think the vent pathto EBFS used for drain down of the RCS is used for drawing a bubble

C. **Incorrect**: the PORVs remain closed while drawing a bubble
Plausible: Cycling PORVs would vent the RCS

D. **Incorrect:** The Reactor head vents are used for post accident venting only
Plausible: Cycling head vents would vent the RCS

References: OP 2301D, Filling and Venting the RCS

Student Ref: NONE

Learning Objective:

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR

Comments (Question 34): 08/11/20 DF added a justification for answer "B" to address the NRC's review comments on 7/24/2020.

6/23 – changed C to Reactor Head vents cycled to Sample Sink
<<

>>QUESTION 35

The setpoint for TIC-223, LTDN TEMP CNTL (C-02) is changed from 120 ºF to 115 ºF.

How does the affected RBCCW header temperature (C-05) respond?

Facility ___(1)___ temperature stabilizes at ___(2)___ temperature.

	(1)	(2)
A.	One	a lower
B.	One	the same
C.	Two	a lower
D.	Two	the same

<<

>>QUESTION 35

K&A Rating: 008 A1.02 (RO 2.9)

K&A Statement: 008 Component Cooling Water System (CCWS), A1.02 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: CCW temperature.

Key Answer: **D**

Justification:

A. Incorrect: Letdown heat exchanger is cooled by facility 2 RBCCW
Plausible: Might think Facility 1 RBCCW cools L/D Hx

B. Incorrect: Letdown heat exchanger is cooled by facility 2 RBCCW
Plausible: Might think Facility 1 RBCCW cools L/D Hx

C. Incorrect: RBCCW header temperature will not change appreciably
Plausible: LD Hx outlet temperature will increase

D. CORRECT: Facility 2 temperature will remain the same, the RBCCW Hx TCV will modulate to maintain a constant temperature.

References: None

Student Ref: NONE

Learning Objective:

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 35): 09/09/2020 jwr. No change to question. Don Jackson questioned if there were two correct answers. There isn't because the stem has "How does the affected …."

Changed A & C to "lower" vice "higher" (Herb and Pete)
<<

>>QUESTION 36

What causes annunciator RBCCW SURGE TK AUTO MAKEUP (C-06/7, B-8) to alarm?

A. 2-RB-215-HS is in AUTO and the valve is open for 3 minutes.

B. 2-RB-215-HS is in OPEN and the valve strokes open.

C. 2-RB-215-HS is in OPEN and RBCCW Surge Tank stays below 40% for 3 minutes.

D. 2-RB-215-HS is in AUTO and the second PMW pump starts.

<<

>> QUESTION 36

K&A Rating: 008 K4.02 (RO 2.9)

K&A Statement: 008 Component Cooling Water System (CCWS); K4.02 Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the surge tank, including the associated valves and controls

Key Answer: **A**

Justification:

A. CORRECT: 2-RB-215 HS is in AUTO and the valve opens for 3 minutes

B. Incorrect: Valve needs to open for 3 minutes and be in "AUTO".
Plausible: Could forget the HS must be in "AUTO" and be open for 3 minutes.

C. Incorrect: 2-RB-215 HS needs to be in AUTO for alarm to sound
Plausible: might think 3 minutes is trigger for alarm

D. Incorrect: PMW not part of alarm circuit
Plausible: might confuse second PMW pump as part of circuit

References: ARP 2590E-046
25203-32015 sh.7

Student Ref: NONE

Learning Objective: 281948 (MB-03014) Describe the functions of the following RBCCW System Control Room controls, including how controlled components are affected by each mode or position of the controls:
E) RBCCW Surge Tank Makeup Valve

Question Source: New

Question History: Modified (ID 6000018 from Unused Exam Bank)

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 36): 02SEP20 DF Changed B from HS in AUTO to HS in OPEN based on comments from review by Don Jackson. John updated "B" answer justification on 09/09/2020.
<<

>>QUESTION 37

The plant is operating at 100% power.

The selected Pressurizer pressure controller fails to 100% output.

Reactor Coolant System pressure will …

A. lower until the reactor trips on TM/LP.

B. lower, then stabilize at ~2200 psia.

C. rise, then stabilize at ~2350 psia.

D. rise until the reactor trips on High Pressure.

<<

>>QUESTION 37

K&A Rating: 010 K3.01 (RO 3.8)

K&A Statement: 010 Pressurizer Pressure Control System (PZR PCS), K3.01 Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: RCS.

Key Answer: **A**

Justification:
A. CORRECT: Controller fails high causes the PZR PCS to sense a high pressure condition. Spray valves open and proportional heaters go to minimum output. B/U heaters energize at ~2200 psia but won't overcome the effect of the spray valves. Plant will eventually trip on TM/LP

B. Incorrect: PZR PCS senses a high pressure condition, trips on TM/LP.
Plausible: might think B/U heaters will recover pressure

C. Incorrect: PZR PCS senses a high pressure condition and will act to lower press.
Plausible might think PZR PCS will act to raise pressure until B/U heaters trip.

D. Incorrect: PZR PCS senses a high pressure condition and will act to lower press.
Plausible: Student might forget about the 2350 psia cutoff

References: PLC-01-C Lesson Plan

Student Ref: NONE

Learning Objective:

Question Source: New

Question History: Modified from South Texas Project NRC Exam 2018 (#32)

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 37):
<<

>>QUESTION 38

The plant is operating at 100% power.

The selected transmitter for Pressurizer pressure control, PT-100X, fails low.

The Pressurizer pressure control response would include …

	proportional heaters	back up heaters
A.	maximum	off
B.	minimum	on
C.	minimum	off
D.	maximum	on

<<

>>QUESTION 38

K&A Rating: 010 K6.01 (RO 2.7)

K&A Statement: 010 Pressurizer Pressure Control System (PZR PCS), K6.01 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems

Key Answer: **D**

Justification: The pressurizer Pressure Control system will sense a **low pressure** condition when the transmitter fails low. The proportional heaters will go to maximum and the B/U heaters will energize in response to the low pressure condition.

A. **Incorrect:** B/U heaters will energize
Plausible: might think confuse B/U heaters don't energize on low pressure

B. **Incorrect:** Proportional heaters go to maximum
Plausible: might think sensed pressure is higher than actual

C. **Incorrect**: Proportional heaters go to maximum, B/U heaters energize
Plausible: might think sensed pressure is higher than actual

D. **CORRECT:** PZR PCS senses a low pressure condition. Spray valves will close, proportional heaters go to maximum output and B/U heaters energize (xmitted pressure < 2200 psia)

References: PLC-01-C Lesson Plan
25203-29199 sh5, PLPCS schematic
Student Ref: NONE

Learning Objective: 281907 (MB-02982) Given the plant with a steam bubble in the pressurizer, and given a pressurizer level or pressure transmitter failure (high or low) on either control channel (selected or non-selected), describe:

a. The system response that would result in this failure\
b. The actions necessary to mitigate this failure
c. The plant response if no operator actions are taken

Question Source: modified
Question History: 4003800 (2004 ILT NRC exam)
Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 38): removed column with spray valve response due to NRC comments DF 11AUG20

<<

>>QUESTION 39

Given a Reactor Protection System (RPS) trip select the correct basis for the trip?

	Trip	Basis
A.	Loss of Turbine.	RCS protection against overpressurization.
B.	Variable High Power – High.	Protection against fuel centerline melt.
C.	Reactor Coolant Flow – Low.	Protection against Departure from Nucleate Boiling.
D.	Steam Generator Pressure - Low.	Ensures Reactor Trip coincident with an ESF Actuation.

<<

>>QUESTION 39

K&A Rating: 012G2.2.25 (3.2, 4.2)

K&A Statement: Reactor Protection System. Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Key Answer: **C**

Justification:

A. **Incorrect:** The Loss of Turbine trip basis states that it provides Turbine protection, reduces the severity of ensuing transients and helps avoid the lifting of the main steam line safety valves during ensuing transient, thus extending the service life of these valves. The Pressurizer pressure - High trip basis is to provide RCS protection against overpressurization in the event of a loss of load without a reactor trip.
Plausible: A loss of the Turbine will result in RCS pressure rising, but it is not the basis for this trip.

B. **Incorrect:** The basis for the Variable High Power – High is not fuel centerline melt. It is that it provides reactor core protection against reactivity excursions which are too rapid to be protected by Pressurizer Pressure – High or Thermal Margin/Low Pressure.
Plausible: High power would raise fuel temperature.

C. **Correct**: The Reactor Coolant Flow – Low basis states that the trip provides core protection to prevent Departure fron Nucleate Boiling (DNB) in the event of a sudden significant decrease in reactor coolant flow.

D. **Incorrect:** The Steam Generator Pressure - Low basis states that the trip provides protection against excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant.
Plausible: The basis is similar ro the Containment Pressure – High trip which is to provide assurance that a reactor trip is initiated with a safety injection. Similar in that Steam Generator low pressure will initiate a reactor trip and an ESF actuation (Main Steam Isolation). Like a Containment pressure – High will initiate a reactor trip on high containment pressure and an ESF actuation (Safey Injection Actuation).

References: RPS-01-C Rev 7, Technical Specification Bases including Chg No. 397
Student Ref: NONE

Learning Objective: 282159 (03154-MB) For each Reactor Protection System reactor trip, describe the Technical Specification basis and required trip setting.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.5 / 41.7 / 43.2

Comments (Question 39): 6/2/20 jwr. The second validation with PF identified that the question was more than was needed in the K/A. The question was changed to remove the third column (parameter) and two of the RPS trips were changed to get plausible answers. 5/8/20 jwr changed “input” to “parameter”. OPS comment on validation.
<<

>>QUESTION 40

What is the effect of de-energizing VA-10 on 2-FW-51A, #1 Main Feedwater regulating Valve?

Valve 2-FW-51A will…

A. ONLY respond to a Facility 1 MSI signal.

B. ONLY respond to a Facility 2 MSI signal.

C. respond to EITHER a Facility 1 OR Facility 2 MSI signal.

D. NOT respond to ANY MSI signal.

<<

>>QUESTION 40

K&A Rating: 013 K2.01 (RO 3.6*)

K&A Statement: 013 Engineering Safety Features Actuation System (ESFAS), K2.01 Knowledge of bus power supplies to the following: ESFAS/safeguards equipment control

Key Answer: **D**

Justification:

A. **Incorrect:** A loss of VA-10 will de-energize Actuation Cabinet 5 so a MSI signal won't be sent to #1 FRV.
Plausible: might forget power supply to ACT-5

B. **Incorrect:** A Facility 2 MSI signal will be sent to the FRV but the valve will be locked up
Plausible: might forget the power supply to the lock up solenoids

C. **Incorrect**: Facility 1 won't send a signal to the FRV, which is locked up anyway
Plausible: With no issues, this would be the correct answer.

D. **CORRECT:** VA-10 powers the Lock-Up solenoids for 2-FW-51A, a loss of VA-10 will lock-up the FRV

References: AOP 2504C, Loss of VA-10

Student Ref: NONE

Learning Objective: 281375 (MB-02469) 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

Question Source: Bank (56301)

Question History: 2000 NRC Exam Q#87 (MB-02666)

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 40):
<<

>>QUESTION 41

The plant is MODE 5 on Shutdown Cooling (SDC). Two SDC heat exchangers (HX) are in service with 2000 gpm to each.

- “A” RBCCW header flow is 6,500 gpm
- “B” RBCCW header flow is 6,600 gpm
- Opening or closing a CAR Emergency Outlet valve will change flow by 1,500 gpm
- Opening or closing a CAR Normal Outlet valve will change flow by 500 gpm

The “A” SDC HX will be removed from service with RBCCW isolated to the HX. The “B” SDC HX RBCCW flow will be raised by 2,000 gpm.

What operation of a CAR valve is necessary to prevent exceeding procedure limits?

A. Opening the “A” CAR Emergency Outlet valve 2-RB-28.3A.

B. Closing the “B” CAR Emergency Outlet valve 2-RB-28.3B.

C Opening the “C” CAR Normal Outlet valve 2-RB-28.2C.

D. Closing the “D” CAR Normal Outlet valve 2-RB-28.2D.

<<

>>QUESTION 41

K&A Rating: 022A1.04 (3.2)

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

Key Answer: **B**

Justification:

A. Incorrect: Opening the "A" CAR Emergency Outlet valve 2-RB-28.3A is not necessary to prevent exceeding procedural limits. The applicable procedural limit is 4,000 gpm RBCCW header flow. The bases is that less than 4,000 gpm could lift RBCCW relief valves. Flow was lowered to 4,500 gpm (6,500 – 2,000 = 4,500). "A" RBCCW header flow is still above 4,000 gpm even when the RBCCW flow is isolated to the "A" SDC. Therefore adding RBCCW flow to the "A" RBCCW header is not necessary to protect against lifting RBCCW relief valves.
Plausible: Examinee has to know the RBCCW header flow value (< 4,000 gpm) which will challenge lifting relief valves.

B. Correct: Closing the "B" CAR Emergency Outlet valve 2-RB-28.3B is correct. Adding 2,000 gpm of flow to the "B" RBCCW header will bring total header flow to 8,600 gpm (6,600 + 2,000 = 8,600 gpm). This is above the pump runout flow of 8,000 gpm. Therefore closing the "B" CAR emergency outlet valve is required to stay within procedural limits. Reducing flow by 1,500 gpm will lower "B" RBCCW header flow to 7100 gpm.

C. Incorrect: Opening the "C" CAR Normal Outlet valve 2-RB-28.2C is not correct. Opening this valve will not restore flow to above the procedural limit of 5,500 gpm. The bases of the 5,500 gpm limit is that at lower than 5,500 gpm RBCCW pump vibration occurs. Opening the "C" CAR normal outlet valve will add 500 gpm to header flow but this is not enough to bring "A" RBCCW header flow to 5,500 gpm (6,500 – 2,000 = 4,500, then 4,500 + 500 = 5,000).
Plausible: Examinee has to know the RBCCW header flow value (<5,500 gpm) which will result in higher RBCCW pump vibration.

D. Incorrect: Closing the "D" CAR Normal Outlet valve 2-RB-28.2D is not correct. Closing this valve will not lower flow to less than the procedural limit of 8,000 gpm. Closing the "D" CAR normal outlet valve will not lower the "B" RBCCW header flow to less than 8,000 gpm (6,600 + 2,000 = 8,600, then 8,600 – 500 = 8,100).
Plausible: Examinee has to know the RBCCW header flow value (>8,000 gpm) which will result in pump runout.

Note: CAR Emergency and Normal Outlet valve flow values are provided because initial license candidates would not readily have this knowledge. Without this information the question would have a LOD of 5.

References: OP 2330A Rev 028-00, OP 2310 Rev 032, Student Ref: NONE

Learning Objective: 281950 (0319-MB) State the basis or reason for each RBCCW System precaution, caution, or note contained in OP 2330A.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.5

Comments (Question 41): 08/19/2020 jwr. Changed question due to validator comment. Stem changed from “B” header flow of 6,400 gpm to 6,600 gpm. The comment was that answer “D” is also correct because it is acceptable to be above 7,300 gpm, and we often are above this, it is just better to be less than 7,300 gpm.

08/07/2020 jwr. Completed changes to stem,answers, and justification requested by the NRC. The NRC identified the stem could be written so it is easier to understand, that the answers did not need a second part (could just put second part in justification), and that the justification should include why the valve flows were needed to be provided. They provided a rewording of the stem question that we used for the rewrite.

6/21/20 jwr. Comment from John W that emergency valves are 1,500 gpm not 2,000 gpm. Made change in stem to emergency valves are 1,500 gpm. Checked answers still work and they do.
5/19/20 jwr changed “A” header flow to 6,500 gpm to move a little further away from the < 4,000 gpm limit which protects against challenging lifting RBCCW relief valves. Specified normal and emergency valve flows. Changed due to first OPS validation comments.
<<

>>QUESTION 42

The plant experienced a Large Break Loss of Coolant Accident.

All requirements for terminating Containment Spray have been met.

In accordance with EOP 2532, Loss of Coolant Accident, how is the Containment Spray Actuation Signal (CSAS) reset so that the Containment Spray (CS) system is available to automatically respond to any future CS actuation signals?

A. Place the CS pump handswitches to STOP, and then reset CSAS at ESAS.

B. OVERRIDE the CSAS start signals and stop the CS pumps, then reset CSAS at ESAS.

C. Reset CSAS at ESAS; then OVERRIDE the CSAS start signals and stop the CS pumps.

D. Reset CSAS at ESAS, and then place the CS pump handswitches to STOP.

<<

>>QUESTION 42

K&A Rating: 026A4.05 (RO 3.5)

K&A Statement: 026 Containment Spray System (CSS), A4.02 Ability to manually operate or monitor in the control room: Containment spray reset switches.

Key Answer: **D**

Justification:

A. **Incorrect:** The pump will not stop with by taking the pump HS to stop when a CSAS is present. The ESAS signal must be reset prior to taking the HS to stop to secure the pumps.
Plausible: Examinee might think the pump is stopped first since this what the students are most familiar with. Students are familiar with overriding which is taking the HS to start to clear the CSAS signal and then taking the HS to stop which then stops the pump.

B. **Incorrect**: Overriding the CSAS start signals then securing the CS pumps is not correct. The procedure specifies reseting at ESAS first then stopping the CS pumps.

Plausible: Examinee might think the pump is stopped first since this what the students are most familiar with. Students are familiar with overriding which is taking the HS to start to clear the CSAS signal and then taking the HS to stop which then stops the pump.

C. **Incorrect:** Once the ESAS signal is reset, the pump can be stopped w/o overriding.
Plausible: Examinee might think the signal still needs to be reset even after the signal is reset at ESAS since this is what they are most famliar with.

D. **CORRECT:** Once the ESAS signal is reset, the pump can be stopped w/o override, i.e. simply taking the CS HS to stop.

References: EOP 2532, Loss of Coolant Accident
EOP 2541, APP 25, resetting ESAS

Student Ref: NONE

Learning Objective: 281142 (MB-02322) Describe the functions of the following Containment Spray System Control Room controls, including how the controlled component(s) is/are affected by each mode or position of the control

a) Containment Spray Pump A and B control switches

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 42):

02SEP20 DF – added ‘In accordance with EOP 2532…’ based on comments from Don Jackson review. Changed ‘secure’ to ‘terminate’ to be in alignment with EOP 2532

8/20/20 DF - Restored question to previous revision as requested by NRC. D. Silk felt the examinees should be knowledgeable of how a safety signal is reset even though the process is not reached until deep into the procedure.

8/17/20 DF - Rewrote question due to high miss rate (>50%) by validators.

8/11/20 DF - changed “futures” to “future” in stem.

6/30/20 jwr – This was a question missed by ~ 50% on validation; two of the last three validators correctly answered and these changes should enhance the pass rate. Changed stem to make it clear that the CSAS signal is being reset so the CS system is available to automatically respond to future CSAS signals. Changed answers to make wording consistent among answers. Previously DF changed stem to ”Procedurally, how is the Containment Spray Actuation Signal reset” and on 6/22/20 DF -Changed B & C to read “ OVERRIDE the CSAS start signals”.

<<

>>QUESTION 43

The plant is holding power at 47% in order to start a second Steam Generator Feed Pump.

A reactor trip occurs.

How do the Atmospheric Dump Valves (ADVs) and Condenser Steam Dumps respond?

	ADVs	Condenser Steam Dumps
A.	Quick open, then operate on setpoint	Quick open, then operate on setpoint
B.	Quick open, then operate on setpoint	Operate only on setpoint
C.	Operate only on setpoint	Quick open, then operate on setpoint
D.	Operate only on setpoint	Operate only on setpoint

<<

>>QUESTION 43

K&A Rating: 039K1.02 (3.3)

K&A Statement: Main and Reheat Steam System. Knowledge of the physical connections and/or cause-effect relationships between the MRSS and the following systems: Atmospheric relief dump valves.

Key Answer: **D**

Justification:

A. Incorrect: Quick Open not armed with T_{AVE} < 554 °F
Plausible: might forget Quick Open arms > 554 °F

B. Incorrect: Quick Open not armed with T_{AVE} < 554 °F
Plausible: might forget Quick Open arms > 554 °F

C. Incorrect: Quick Open not armed with T_{AVE} < 554 °F
Plausible: might forget Quick Open arms > 554 °F

D. CORRECT: Quick Open not armed with T_{AVE} < 554 °F. T_{AVE} = 554 °F at approximately 60% power. Quick Open not required due to lower Rx Heat production.

References: RRS-01-C, Reactor Regulating System Lesson Plan
Student Ref: NONE

Learning Objective: 282174 (MB-03167) Given any of the following events or scenarios, describe the response of the Atmospheric and Condenser Steam Dump Valve Control System:
A. Turbine trip with T_{AVE} anywhere in the normal operating range

Question Source: Modified

Question History: 451461 (MB-0155746) NRC 2000 Q94

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.4

Comments (Question 43)::

<<

>>QUESTION 44

The plant is operating at 100% power.

The #1 Atmospheric Dump Valve fails open to 25%.

How will #1 S/G level respond? Level will ___(1)___, then ___(2)___.

	(1)	(2)
A.	rise due to S/G swell	stabilize at a level greater than setpoint
B.	lower due to S/G shrink	return to setpoint and stabilize
C.	rise due to feed flow/steam flow mismatch	return to setpoint and stabilize
D.	lower due to feed flow/steam flow mismatch	stabilize at a level lower than setpoint

<<

>>QUESTION 44

K&A Rating: 059A3.02

K&A Statement: 059 Main Feedwater (MFW) System A3.02 Ability to monitor automatic operation of the MFW, including: programmed levels of the S/G.

Key Answer: **D**

Justification:

A. Incorrect: Level will no stabilize higher than setpoint
Plausible: There is an very small initial swell when valve fails open

B. Incorrect: Level will lower but not due to shrink
Plausible: reasonable to assume level lowers due to shrink

C. Incorrect: Level will not rise. Steam flow > Feed flow = Level lowers
Plausible: might think feed flow is greater than steam flow (steam robbed from turbine)

D. CORRECT: Steam flow is greater than feed flow so level lowers. Level will be essentially stable (level error will try to recover)

References: Lesson Plan FWC-01-C, Feedwater Control System

Student Ref: NONE

Learning Objective:

Question Source: Mod?

Question History: Farley 2012 NRC Q44

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.5 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 44):

<<

>>QUESTION 45

Millstone Unit 2 was operating at 100% when a loss of Bus DV-10 occurred. The plant tripped. Given the following conditions;

- DV-10 remains deenergized.
- EOP 2541, Appendix 4 Followup Actions have been completed.
- The Auxiliary Feedwater OVERRIDE/MAN/START/RESET switches have been taken to RESET.

What is the status of the Auxiliary Feedwater Regulating valves (AFRV) 2-FW-43A and 2-FW-43B?

A. Both AFRVs can be controlled from the control room.

B. Neither AFRV can be controlled from the control room.

C. Only AFRV 2-FW-43A can be controlled from the control room.

D. Only AFRV 2-FW-43B can be controlled from the control room.

<<

>>QUESTION 45

K&A Rating: 061K6.01

K&A Statement: 061 Auxiliary/Emergency Feedwater (AFW) System. K6.01 Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners

Key Answer: **D**

Justification:

A. **Incorrect:** Both Auxiliary Feedwater Regulating Valves (AFRV) can not be controlled from the control room. The loss of DV-10 will cause AFRV 2-FW-43A to fail open and control from the control room will be lost.
Plausible: Examinee must understand both what the loss of DV-10 does and what the Auto Aux. Feed Actuation signal does and the affect of resetting the signal on the AFRVs.

B. **Incorrect:** Neither AFRV can be controlled from the control room is not correct. AFRV 2-FW-43B can be controlled from the control room.
Plausible: Examinee must understand both what the loss of DV-10 does and what the Auto Aux. Feed Actuation signal does and the affect of resetting the signal on the AFRVs.

C. **Incorrect**: AFRV 2-FW-43A can not be controlled from the control room and 2-FW-43B can be controlled from the control room.
Plausible: Examinee must understand both what the loss of DV-10 does and what the Auto Aux. Feed Actuation signal does and the affect of resetting the signal on the AFRVs.

D. **CORRECT:** Only AFRV 2-FW-43B can be controlled from the control room. The loss of DV-10 does not affect 2-FW-43B but fails open 2-FW-43A.

References: AOP 2506A Loss of Vital 125 VDC Instrument Panel DV-10 rev. 005, EOP 2541 Appendix 4 Followup Actions, rev. 007-00 Student Ref: NONE

Learning Objective: 280901 (02156-MB) Given a loss of DC control power to the following components, predict the response of the AFW system and identify correct operator response to mitigate the event: A) Turbine Driven Auxiliary Feedwater Pump Controls, B) AFW-FCVs, FW-43A and 43B.

Question Source: 5000059-MB (452692) Modified

Question History: 2005 NRC, Question 55, 061K6.01, Comprehensive

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility

Comments (Question 45): 08/05/2020 jwr. Replaced K/A and question as a result of NRC feedback on the exam. Feedback was hard to write a question on the effect of the loss of an AFW pump. Essentially a new question. Changed stem and all answers.
6/21/20 jwr. Change during test to stem to specify pump was steam bound. Comment from John W that this is needed and helpful.
DF - Added “due to leak by through discharge check valves 2-FW-12A and 2-FW-8A” to stem. 6/22 removed “and feed to the #1 S/G”, changed C to “vent for 30 minutes”.
<<

>>QUESTION 46

The unit entered AOP 2580, Degraded Voltage.

Safety Bus parameters are as follows:

- Voltage = 3950 VAC.
- Frequency = 60 Hz.

Why are in-progress surveillances of safety related pumps terminated?

A. To ensure the pumps are available for the impending reactor trip.

B. Overheating of the motor cabling and windings could occur.

C. Pump motor speed lowers, resulting in insufficient pump ΔP.

D. Pump motor current lowers, resulting in insufficient pump ΔP.

<<

>>QUESTION 46

K&A Rating: 062A2.08 (RO 2.7)

K&A Statement: 062 AC Electrical Distribution System A2.08 Ability to (a) predict the impacts of the following malfunctions or operations on the AC distribution system; and (b) based on these predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: consequences of exceeding voltage limitations.

Key Answer: **B**

Justification:

A. Incorrect: Surviellance of a pump does not prevent it from responding to a trip
Plausible: might think a trip is required for AOP 2580 (it isn't)

B. CORRECT: Degrading voltage will cause current to rise to maintain power requirements, the increase in current can cause overheating of motor windings

C. Incorrect: Since frequency hasn't changed, pump speed will remain constant. Especially at the Voltage and Frequency given in the stem
Plausible: might think pump speed lowers causing a lower ΔP.

D. Incorrect: Pump motor current will rise as voltage lowers.
Plausible: might think current lowers thus causing a lower ΔP.

References: AOP 2580, Degraded Voltage

Student Ref: NONE

Learning Objective: 283479 (MB-05530) Determine the effects of a degraded voltage condition on plant components

Question Source: Modified

Question History: 418417 NRC 2000 exam Q79 MB-53588

Cognitive Level: Memory/Fundamental Knowledge: X

Comprehensive/Analysis:

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility

Comments (Question 46): 13AUG20 DF: Changed stem to include bus voltage and frequency. Changed answers C and D to add the effect of insufficient ΔP Based on NRC comments

<<

>>QUESTION 47

The plant tripped from 100% power coincident with annunciator 125VDC LOAD CENTER 201B TROUBLE (C-08, B-21).

The following indications exist on C-08 for Bus 201B:

- BATT AMPS = 0 DC Amperes.
- BATT VOLTS = 132 DC Volts.
- BUS AMPS = 0 DC Amperes.

What is the system configuration of Bus 201B?

A. Breaker D0203, Battery Bus 201B to 125VDC bus 201B, has tripped on overload.

B. Breaker D0202, Battery Charger to Battery Bus 201B, has tripped on overload.

C. DS-1, Battery 201B Fuse, is open (blown).

D. Bus 201B is in a normal post-trip alignment.

<<

>> QUESTION 47

K&A Rating: 063A3.01 (3.7*)

K&A Statement: 063 DC Electrical Distribution A3.01 Ability to monitor automatic operation of the DC electrical system, including: meters, annunciators, dials, recorders, and indicating lights.

Key Answer: **A**

Justification: The vital DC trains consist of 2 busses: the Battery Bus and the 125VDC Bus (201A or B). The busses are tied together by a breaker. The Battery bus consists of the battery and the charger. The 125VDC bus contains all the loads.

A. CORRECT: The plant trip, coincident with 201B trouble, indicates the Battery Bus to125 VDC Bus tie breaker has tripped open. This is supported by 0 amps on the 125VDC Bus.

B. Incorrect: If the Charger breaker trips, the. Battery would be supplying the 125VDC bus so Battery amps would be a negative value and the 125VDC bus amps would not be zero.
Plausible: may think 0 battery amps indicates charger not supplying battery

C. Incorrect: If the battery fuse opens, the battery amps would be zero but the charger would be supplying the 125VDC bus so the 125VDC bus amps would not be zero
Plausible: Battery amps and volts indicate battery disconnect open

D. Incorrect: 201B amps would be indicate other than zero on a trip
Plausible: might think the indications are normal

References: 25203-30024-A

Student Ref: NONE

Learning Objective: 281636 (MB-04875) Given a plant condition, and a sequence of indications, alarms or malfunctions; predict the effect on the following plant systems and/or plant equipment: 125V Vital & Non-vital DC

Question Source: Modified

Question History: 451695 2005 NRC Q59 (MB-5000061)

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 47): Added to justification to better explain the system and reason why C is incorrect per NRC comments. DF 31JUL20

<<

>>QUESTION 48

A natural disaster occurs which places the unit in EOP 2530, Station Blackout. An Extended Loss of All Power (ELAP) is declared.

What operational strategy regarding DC buses and loads is employed?

DC Buses ___(1)___, and DC loads are stripped so that only select ___(2)___ loads remain energized.

	(1)	(2)
A.	remain split	VA-10 & VA-30
B.	remain split	VA-20 & VA-40
C.	are cross-tied	VA-10 & VA-30
D.	are cross-tied	VA-20 & VA-40

<<

>>QUESTION 48

K&A Rating: 063G2.4.20 (RO 3.8)

K&A Statement: 063 DC Electrical Distribution G2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.

Key Answer: **D**

Justification: From EOP 2530, Staion Blackout: NOTE An ELAP is declared when it is predicted that AC power will NOT be restored within ONE hour of the event initiation. This is a judgement call based on existing information, which must be made within FORTY FIVE minutes to allow for DC Bus load stripping.

From FSG-04, ELAP, To conserve Battery Life, this strategy will leave only selected loads on 120 VAC Vital Panels VA20 and VA 40 energized, until 480 VAC power is restored to power the 201B Battery Charger.
The following steps will cross-tie DC Busses 201A and 201B and their respective batteries.

A. Incorrect: DC buses are cross-tied and all Facility 1 loads are de-energized
Plausible: Might think Facility 1 loads are used for monitoring plant parameters.

B. Incorrect: DC buses are cross-tied, select VA-20 & VA-40 loads remain energized
Plausible: Might think DC buses shouldn't be ever be cross-tied during an ELAP

C. Incorrect: DC buses are cross-tied and all Facility 1 loads are de-energized
Plausible: Might think Facility 1 loads are used for monitoring plant parameters

D. CORRECT: DC buses are cross-tied, select VA-20 & VA-40 loads remain energized

References: EOP 2530, Station Blackout
EOP 25 FSG-04, ELAP DC Bus Load Shed/Management
FLEX one page study guide

Student Ref: NONE

Learning Objective: None found

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility

Comments (Question 48): Provided better justifications, included the note in the justification, reworded the stem, per NRC comments. DF 31JULY20

<<

>>QUESTION 49

The plant is at 12% power. Bus 24E is aligned to Bus 24D.

A fault occurs on Bus 24C and causes the RSST supply to 24C (A302) to trip open.

Which one of the following operator actions is required per the applicable procedure, to prevent damage to the "A" EDG?

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

B. Align the "B" Service Water header to the "A" EDG.

C. Depress the "A" EDG Emergency Stop pushbuttons.

D. Place the "B" SW pump handswitch in the START position.

<<

>> QUESTION 49

K&A Rating: 064K3.03 (RO 3.6)

K&A Statement: 064 Emergency Diesel Generator(ED/G) K3.03 Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: ED/G (manual loads)

Key Answer: **C**

Justification:

A. Incorrect: EOP 2525 SPTA directs that IF the diesel generator output breaker can not be closed, THEN TRIP the Diesel Generator, not perform a normal shutdown.
Plausible: With the EDG running unloaded there is enough time to shut it down normall because the heat load is low. Normal shutdown is generally preferable because it is less stressful to engine.

B. Incorrect: There is no procedural guidance provided to allow crosstie of Facility 2 Service Water with the Facility 1 EDG.
Plausible: It is physically possible to align Facility 2 Service Water to the Facility 1 EDG.

C. CORRECT: The question describes a fault that divorces Bus 24C from its RSST source. Bus 24E aligned to Facility 2 Bus 24D indicates that swing bus 24E is powered from Facility 2. Since the "A" EDG is running without any service water, it should be tripped to prevent damaging the machine. The fault on Bus 24C will prevent the “A” EDG breaker from closing. The “B” SW pump is not aligned to Facility 1. EOP 2525 SPTA directs that IF the diesel generator output breaker can not be closed, THEN TRIP the Diesel Generator. The "A" EDG emergency trip pushbuttons will be used because of the LNP auto start signal.

D. Incorrect: The "B" Service pump is aligned to Facility 2. And even if you could get Service Water to the “A” EDG with a fault of 24C it will not be able to close on to the bus.
Plausible: Starting the Standby “B” Service Water pump even when it is not electrically or mechanically aligned to replace the “A” or “C” is directed in the loss of Service Water procedure.

References: EOP 2525 SPTAs, EDG Lesson Text

Student Ref: NONE

Learning Objective: 283649 (05424-MB) Outline the instructions and contingency actions for immediate actions in EOP 2525, Standard Post Trip Actions.

Question Source: Bank

Question History: 0053401-MB (310051, 453482, 453482) 2014 NRC exam Q48. Enhanced justification and changed stem from Mode 3 to Mode 2 to ensure the reactor is tripped and EOP 2525 is the appropriate guidance.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.10 Administrative, normal, and emergency operating procedures for the facility.

Comments (Question 49): 8/19/20 jwr. Change stem from 10% to 12% to help examinees not miss read 10% as 100%.

6/30/20 jwr – replaced question. Low success rate in validation. Prior to reoplacement added “to override the LNP signal” to C & D.

>>QUESTION 50

A Rapid Downpower is in progress due to a Steam Generator Tube Leak (SGTL).

During the downpower, the SGTL increases.

- RM-5099, SJAE Radiation Monitor reaches its HIGH alarm setpoint.
- The plant is currently at 70% power.

Describe the expected response of RM-4262, Steam Generator Blowdown Rad Monitor as the Rapid Downpower continues.

A. Readings are not predictable due to blowdown being isolated.

B. Readings are not predictable due to lowering power level from 100% to 70%.

C. Readings will rise due to the higher leak rate.

D. Readings will lower due to lowering power.

<<

>>QUESTION 50

K&A Rating: 073A1.01(RO 3.2)

K&A Statement: 073 Process Radiation Monitoring (PRM) System A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the PRM systems controls including: radiation levels

Key Answer: **A**

Justification:

A. CORRECT: When RM-5099 goes into HIGH alarm, Blowdown isolates, securing process flow to RM-4262. The rad monitor reading is no longer representative of S/G activity.

B. Incorrect: The process flow to RM-4262 is secured.
Plausible: reasonable to assume power and leak rate offset each other.

C. Incorrect: The process flow to RM-4262 is secured.
Plausible: readings would rise due to higher leak rate if RM was sampling.

D. Incorrect: The process flow to RM-4262 is secured.
Plausible: If isolation is not considered, might think reading will decrease with power.

References: ARP 2590H-037, RI-5099 STEAM JET AIR EJECTOR

Student Ref: NONE

Learning Objective: 282110 (MB03134) Describe the automatic actions and logic associated with the SJAE and/or S/G Blowdown Radiation Monitoring Subsystems

Question Source: Modified

Question History: 451923 2014 NRC Q49 (MB-2014026)

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.5 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 50): changed stem to include current power level, modified B to state power levels per NRC comments. DF 31JULY20

<<

>>QUESTION 51

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

- A Steam Generator Tube Rupture in the #2 SG.
- RM-4299C, Main Steam Line #2 radiation monitor went into alarm.
- The crew tripped the reactor and performed EOP 2525 Standard Post Trip Actions.
- Both S/G levels are now ~ 30% and rising.

Which one of the following describes the response of RM-4299C following the trip, assuming no fuel damage has occurred?

A. Lowers towards normal due to the effect of the reactor trip.

B. Lowers towards normal due to a lower primary to secondary D/P.

C. Remains at the alarm value until SG #2 is isolated.

D. Remains at the alarm value until SG #2 NR level rises above 40%.

<<

>>QUESTION 51

K&A Rating: 073K5.01 (2.5)

K&A Statement: Process Radiation Monitoring (PRM) System. Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: Radiation theory, including sources, types, units, and effects

Key Answer: **A**

Justification:

A. **CORRECT**: Lowers towards normal due to the effect of the reactor trip is correct. The Main Steam Line Radiation Monitor (MSLRM) RM-4299C detects N-16. N-16 is dependent on the power level and goes away almost immediately following the reactor trip.

B. **Incorrect:** The MSLRM lowers towards normal as a result of the lowering N-16 production. And the N-16 production lowers due to the lower reactor power level not due to the lowering DP between the Steam Generator (SG) and RCS.
Plausible: The N-16 level will lower as the DP between the SG and RCS lowers, since the lleak rate lowers. However the leak rate would still be significant and would not return to a normal pre rupture level.

C. **Incorrect**: N-16 production will drop following reactor shutdown and before the SG is isolated. Isolating the SG will not affect the reading on the radiation monitor.
Plausible: Isolating the SG indicates the leakage as been stopped so this appears reasonable. But the isolation is from the SG to the environment not the RCS to the SG.

D. **Incorrect**: Raising level to 40% is used for iodine scrubbing and will not affect N-16 production.
Plausible: Plant procedures restore and maintain level in the ruptured SG at 40-45%. This is done to limit radiation release. The examinee could think it is lowering N-16.

References: RMS-00-C Radiation Monitor System Lesson Text rev. 7 chg 8

Student Ref: None

Learning Objective: 282114 (06436-MB) Describe the N-16 Radiation Monitor response to detection of primary-to-secondary leakage.

Question Source: Bank 2014027-MB (451924, 453024)

Question History: Millstone 2014 NRC Exam Q50

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.5

Comments (Question 51): 8/5/2020 jwr. K/A replacement and new question as a result of NRC feedback on 7/24/2020. The NRC's comment for the question submitted was that it was at the General Employee level. The question was a direct match with the K/A but at too low a level. The new K/A was provided to raise the level of difficulty.
<<

>>QUESTION 52

The plant is operating at 100% power. Intake temperature is 50°F

A Facility 1 Loss of Normal Power (LNP) occurs.

How will the Service Water system flow rates respond?

	Facility 1	Facility 2
A.	Lower flow	Same flow
B.	Lower flow	Higher flow
C.	Higher flow	Same flow
D.	Higher flow	Higher flow

<<

>>QUESTION 52

K&A Rating: 076A3.02 (RO 3.7)

K&A Statement: 076 Service Water System (SWS) A3.02 Ability to monitor automatic operation of the SWS, including: emergency heat loads.

Key Answer: **B**

Justification: During an LNP actuation, the affected Facility's SW to the TBCCW heat exchangers will isolate. Also the affected Facility's EDG SW bypass valve will close and the EDG SW supply valve will open (this will be a net decrease in SW flow). These configuration changes will result in a net decrease in the affected Facility's SW flow. The unaffected Facility will respond by picking up TBCCW heat loads once carried by the affected Facility, resulting in increasing SW flow.

A. Incorrect: Facility 2 will increase due to cooling all TBCCW loads
Plausible: Might forget that SW to TBCCW isolates on LNP signal

B. CORRECT: LNP will cause SW flow to lower due to EDG bypass valve closing and SW-3.2B closing. Facility 2 flow will increase due to higher flow through TBCCW Hx's

C. Incorrect: SW-3.2B closes and EDG bypass valve closes > lower Facility 1 flow
Plausible: might forget 3.2B closes and think EDG flow increases

D. Incorrect: SW-3.2B closes and EDG bypass valve closes > lower Facility 1 flow
Plausible: might forget 3.2B closes and think EDG flow increases

References: None
Student Ref: NONE

Learning Objective:
Question Source: New
Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR41.5 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 52): 09/09/2020. Don Jackson thought there was a time element and there isn't. Left questions "as-is". Validated on simulator DF 5/19/2020. Expanded justification due to NRC comments. DF 31JUL20.
<<

>>QUESTION 53

What valve receives an open signal during a SIAS?

A. 2-SW-3.1A, ‘A’ Service Water Header to RBCCW Heat Exchangers.

B. 2-SW-3.2A, ‘B’ Service Water Header to TBCCW Heat Exchangers.

C. 2-RB-8.1A, SFPC HX 20A RBCCW Header ‘A’ Outlet Valve.

D. 2-SW-8.1A, ‘A’ RBCCW Heat Exchanger Temperature Control.

<<

>>QUESTION 53

K&A Rating: 076K4.03 (RO 2.9*)

K&A Statement: 076 Service Water System (SWS) K4.03 Knowledge of SWS design feature(s) and/or interlock(s) which provide for the following: Automatic opening features associated with SWS isolation valves to CCW heat exchangers

Key Answer: **D**

Justification:

A. Incorrect: Valve doesn't receive any signal on SIAS
Plausible: Might think valve gets a signal even though it's always open

B. Incorrect: Valve receives a closed signal on SIAS
Plausible: might think valve gets an open signal

C. Incorrect: Valve receives a closed signal on SIAS
Plausible: an open signal would provide cooling

D. CORRECT: valve opens on SIAS

References: 25203-26008

Student Ref: NONE

Learning Objective: 281380 (MB-02476) Describe each of the 11 ESAS actuations or trips with respect to:
a) Conditions sensed and setpoints
b) Purpose
c) Functions

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 53): modified plausibility for C (answer was changed without changing the justification) in response to NRC comments. DF 3AUG20

<<

>>QUESTION 54

What would be the effect of de-energizing bus DV-20 on the Main Steam Isolation Valves (MSIVs)?

A. Neither valve closes.

B. Only #1 MSIV closes.

C. Only #2 MSIV closes.

D. Both MSIVs close.

<<

>>QUESTION 54

K&A Rating: 078K1.05 (RO 3.4*)

K&A Statement: 078 Instrument Air System (IAS) K1.05 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: MSIV air

Key Answer: **D**

Justification: Both MSIVs are supplied air through 4 solenoid valves (2 parallel paths with 2 solenoids in series in each path). Two of the solenoids are powered from DV-10 and two from DV-20. If any solenoid de-energizes, the air is vented from the MSIV, closing it. If either bus DV-10 OR DV-20 de-energizes, BOTH MSIVs will go close.

A. Incorrect: see above
Plausible: Might think the MSIV solenoids are powered from AC sources

B. Incorrect: see above
Plausible: might think DV-20 powers opposite train MSIV solenoid

C. Incorrect: see above
Plausible: might think DV-20 only powers #2 MSIV solenoid

D. CORRECT: De-energizing any of the Instrument Air SOVs will close the MSIV

References: 25203-28202, MSIV air logic diagram

Student Ref: NONE

Learning Objective: 281716 (MB-02734) Given a set of plant conditions and a Main Steam Isolation Actuation, describe the response of the MS System components

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X

Comprehensive/Analysis:

10CFR55: 10CFR41.7 Design, components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Comments (Question 54): Changed question to give # of SOVs, candidate needs to know how many SOVs to close valve. 6/23 Changed to a 2x2 with only 1 and 4 being options. Change question to address the cause/effect of losing a power supply to solenoids providing air to MSIV in response to NRC comments. DF 03AUG20 <<

>>QUESTION 55

The plant is performing OP 2209A, Refueling Operations.

- #1 & #2 S/G Secondary sides (hand-holes and manways) are open.
- #1 & #2 S/G Safety Valves (MSSVs) are removed for testing.

Technical Specifications are being complied with when …

A. the penetration is capable of being closed under administrative control.

B. the ventilation system is aligned to maintain Containment at a slight negative pressure.

C. the Reactor Coolant average temperature is maintained less than 200 °F.

D. either the hand-hole job or the MSSVs are being actively worked.

<<

>> QUESTION 55

K&A Rating: 103 G2.2.36 (RO 3.1)

K&A Statement: 103 Containment System G2.2.36 Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.

Key Answer: **A**

Justification:

A. CORRECT: Tech Spec. 3.9.4 (CTMT Penetrations) states the penetrations open to the outside atmosphere are either closed or capable of being closed under administrative control

B. Incorrect: not a TS consideration for CTMT Systems
Plausible: Good idea for outages to keep contamination from traveling out of CTMT

C. Incorrect: This is one reason CTMT Closure is available
Plausible: Might confuse CTMT Closure with CTMT integrity (MODE 1-4)

D. Incorrect: People working the job aren't necessarily the ones setting Closure
Plausible: Might think people at job site are the ones to set Closure

References: TS 3.9.4 Refueling Operations/Containment Penetrations

Student Ref: NONE

Learning Objective: 282686 (MB-06866) Given a set of plant conditions, identify Technical Specification implications concerning Refueling Operations

Question Source: Modified

Question History: 310471 2008 NRC Q54 (MB-0054821)

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR55.41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments (Question 55): Changed stem to be specific regarding Tech Spec requirements in response to NRC comments DF 03AUG20.

<<

>>QUESTION 56

The plant is performing OP 2209A, “Refueling Operations”.

- Fuel movement was suspended for 15 minutes.
- AUDIBLE COUNT RATE CHANNEL SELECTOR is selected to ‘A’.
- AUDIBLE COUNT RATE SCALER is selected to ‘X10’.

The WIDE RANGE MONITOR SOURCE RANGE instruments readings before and after suspending fuel movement are as follows:

	before	after
Channel A	79 cps	124 cps
Channel B	77 cps	123 cps
Channel C	70 cps	119 cps
Channel D	72 cps	120 cps

What actions are required by OP 2209A, “Refueling Operations”?

A. Continue to monitor nuclear instruments.

B. Refer to AOP 2558, Emergency Boration.

C. Place the AUDIBLE COUNT RATE CHANNEL SELECTOR to ‘B’.

D. Place the AUDIBLE COUNT RATE SCALER to ‘X100’.

<<

>>QUESTION 56

K&A Rating: 015K5.10 (RO 2.8)

K&A Statement: 015 Nuclear Instrumentation (NIS) K5.10 Knowledge of the operational implications of the following concepts as they apply to the NIS: Ex-core detector operation

Key Answer: **B**

Justification:

A. Incorrect: Counts should not increase without adding fuel assemblies
Plausible: Might think it takes time for counts to reach equilibrium.

B. CORRECT: OP 2209 has the operator refer to AOP 2558 if any unexpected count rate multiplication is indicated

C. Incorrect: Not going to help the unexpected count rate
Plausible: Might want to monitor the highest reading channel

D. Incorrect: Not going to help the unexpected count rate
Plausible: might want to reduce the noise in the Control Room

References: OP 2209A, Refueling Operations, step 4.4.16
Student Ref: NONE

Learning Objective: Describe the response to an unexpected count rate multiplication (i.e., doubling) during Refuel Operations.

Question Source: Bank

Question History: 414113 (MB-06855)

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR55.41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments (Question 56): Changed stem to be specific regarding procedural requirements in response to NRC comments DF 03AUG20

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>>QUESTION 57

Case 1:

- The plant is operating at 100% power.
- Average CET temperature = 554°F.
- T_{HOT} = 594°F.

Case 2:

- The crew is performing a Natural Circulation Cooldown.
- Average CET temperature = 520°F.
- T_{HOT} = 500°F.

In regards to the above, CET indications are...

	Case 1	Case 2
A.	indicating normally.	consistent with natural circulation.
B.	indicating abnormally high.	consistent with natural circulation.
C.	indicating normally.	NOT consistent with natural circulation.
D.	indicating abnormally high.	NOT consistent with natural circulation.

<

>>QUESTION 57

K&A Rating: 017A3.01 (RO 3.6)

K&A Statement: In-Core Temperature Monitor (ITM) A3.01 Ability to monitor automatic operation of the ITM system including: Indications of normal, natural, and interrupted circulation of RCS

Key Answer: **C**

Justification:

A. Incorrect: Natural circulation CET temperatures would be within 10°F of T_{HOT} readings. Plausible: 515°F is a common number in EOPs (Tube rupture C/D). Might think normal CET temperatures are close to T_{COLD}

B. Incorrect: Natural circulation CET temperatures would be within 10°F of T_{HOT} readings. Plausible: 515°F is a common number in EOPs (Tube rupture C/D).

C. CORRECT: The Case 1 CET readings were taken from the PPC (3/23/20). CET temperatures are lower than T_{HOT} and hotter than T_{COLD} since the CETs are located in tubes that 'bypass the core'. Proper natural circulation CET readings would be within 10°F of T_{HOT} readings.

D. Incorrect: CET temperatures are normal for 100% values
Plausible: Might think normal CET temperatures should read close to T_{COLD}

References: EOP 2532, Loss of Coolant Accident, step 39

Student Ref: NONE

Learning Objective:

Question Source: Modified

Question History: Braidwood 2013 Q4

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR55.41.2 General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments (Question 57):Changed Case 1 from too low to too high.

<<

>>QUESTION 58

The plant is operating at 100% power.

Alarm PRESSURIZER CH. X LEVEL HI/LO (C-02/3, A-38) annunciates.

ARP 2590B-213, Pressurizer Channel X Level Hi/Lo, instructs you to use what instrument to confirm whether ACTUAL pressurizer level is rising or lowering?

A. Wide Range Power instrument JI-001.

B. Pressurizer Liquid Space temperature, T101.

C. Letdown flow, F202.

D. Pressurizer pressure safety channel, P102A.

<<

>>QUESTION 58

K&A Rating: 011G2.4.46 (RO 4.2)

K&A Statement: 011 Pressurizer Level Control System (PZR LCS) G2.4.46 Ability to verify that the alarms are consistent with the plant conditions.

Key Answer: **D**

Justification:

A. Incorrect: Wide range power would not differentiate power changes due to pressure
Plausible: might think wide range instruments can show changes in power due to pressure changes.

B. Incorrect: temperature changes are miniscule compared to pressure changes and are delayed (follow pressure changes).
Plausible: Might think temperature change would be a good indicator for PZR level changes

C. Incorrect: Without Charging flow, Letdown is useless
Plausible: Letdown flow is a good indicator if charging flow is available.

D. CORRECT: Actual level tracks with pressurizer pressure. The ARP has the operator check PZR pressure to determine level changes

References: ARP 2590B-213 Pressurizer Channel X Level Hi/Lo
Student Ref: NONE
Learning Objective: 281907 Given the plant with a steam bubble in the pressurizer, and given pressurizer level or pressure transmitter failure (high or low) on either control channel (selected or non-selected), describe:

a) The system response that would result from this failure.
b) The actions necessary to mitigate this failure.
c) The plant response if no operator actions are taken.

Question Source: New
Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR55.41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments (Question 58): 02SEP 20 added ARP to answer based on Don Jackson's comments
6/9/20 Changed B. to Pressurizer Liquid Space temperature, T101.
13AUG20 DF. Changed answer A to WR power from L110X based on feedback from NRC
<<

>>QUESTION 59

The plant is operating at 100% power. Spent Fuel Pool level is lowering.

The crew entered AOP 2578, Loss of Refuel and Spent Fuel Pool Level.

Procedurally, what make-up source is preferred?

A. Boric Acid Storage Tanks (BASTs).

B. Fire Water (FW).

C. Primary Make Up Water (PMW).

D. Refueling Water Storage Tank (RWST).

<<

>>QUESTION 59

K&A Rating: 033A2.03 (RO 3.1)

K&A Statement: Spent Fuel Pool Cooling System (SFPCS) – A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: Abnormal spent fuel pool water level or loss of water level.

Key Answer: **D**

Justification:

A. Incorrect: Not a procedural choice for make-up water.
Plausible: might think this is a good borated source to use.

B. Incorrect: Last choice for make-up water
Plausible: might think AOP is last resort after ARP

C. Incorrect: Second choice for make-up water
Plausible: Normal make-up source for SFP (ARP 2590E-070)

D. CORRECT: Borated sources are considered first to maintain SFP boron concentration > 600 ppm (k_{EFF}<0.95)

References: AOP 2578, Loss of Refuel and Spent Fuel Pool Level.
Student Ref: NONE

Learning Objective:

Question Source: Modified

Question History: Seabrook 2009 Q61

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR55.41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments (Question 59): Changed A from CST to BAST, changed stem from “addressed first” to ‘preferred’ in response to NRC comments. DF 03AUG20

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>>QUESTION 60

Fuel movement is in progress.

The crew has entered AOP 2572, Loss of Shutdown Cooling.

What actions are taken in regards to Fuel movement and Containment Closure?

Fuel movement ___(1)___ and Containment Closure is established ___(2)___.

	(1)	(2)
A.	is stopped	within 8 hours
B.	is stopped	prior to RCS boiling
C.	can continue	within 8 hours
D.	can continue	prior to RCS boiling

<<

>>QUESTION 60

K&A Rating: 034 K1.02 (RO 2.5)

K&A Statement: 034 Fuel Handling Equipment System (FHES)- K1.02 Knowledge of the physical connections and/or cause-effect relationships between the Fuel Handling System and the following systems: RHRS

Key Answer: **B**

Justification:

A. Incorrect: Fuel movement is stopped but closure is set prior to RCS boiling
Plausible: Might think 60 minutes is reasonable amount of time to set closure

B. CORRECT: AOP 2572 stops CORE ALTERATIONS step 1 and requires Closure established prior to RCS boiling

C. Incorrect: CORE ALTERATIONS are stopped; closure is set prior to RCS boiling
Plausible: might think AOP gives latitude to continue to move fuel.

D. Incorrect: CORE ALTERATIONS are stopped.
Plausible: might think AOP gives latitude to continue to move fuel.

References: AOP 2572, Loss of Shutdown Cooling Student Ref: NONE

Learning Objective: 283352 Outline the major actions for a Loss of Shutdown Cooling

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 10CFR55.41.10 Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments (Question 60): Changed 30 minutes to 60 minutes. 6/9/20. 6/22 changed from 60 minutes to 8 hours due to pool being full. Added “What actions are taken in regards to Fuel movement and Containment Closure?” to stem in response to NRC comments 03AUG20.

<<

>>QUESTION 61

The plant is at 100% power.

The Loop 2 Hot Leg temperature (T_{HOT}) input to the Reactor Regulating System (RRS) fails to (615°F).

Which of the following describes the effect of this malfunction?

A. On a plant trip, the Tave controller TIC-4165, will keep the Condenser Dump valves open.

B. Letdown flow will immediately go to the minimum allowed by the Letdown Limiter.

C. On a plant trip, the Quick Open signal from the RRS will not clear (reset).

D. Letdown flow will immediately go to maximum allowed by the Letdown Limiter.

<<

>>QUESTION 61

K&A Rating: 041K6.03 (2.7)

K&A Statement: Steam Dump System (SDS)/Turbine Bypass Control. Knowledge of the effects of a loss or malfunction on the following will have on the SDS: Controller and positioners, including ICS, S/G, CRDS

Key Answer: **A**

Justification:

A. Correct: On a plant trip the Tave controller (TIC-4165) will keep the Condenser Dump valves (CDV) open is correct. The Reactor Regulating System (RRS) Tave will rise and will not lower enough to close the CDVs on the plant trip.

B. Incorrect: Letdown flow will not go to the minimum allowed by the Letdown Limiter. The Loop 2 Th will cause the calculated RRS Tave to rise some but at 100% power the Pressurizer Level setpoint is already at it's maximum value of 65%. And therefore will have no effect on the Pressurizer Level Control System.
Plausible: Letdown flow would go to the minimum allowed by the Letdown Limiter if a cold Leg temperature input failed to it's minimum value of its 100 degree range (515°F) or if the plant was at less than 85% power and the Loop 2 Hot Leg temperature input to the RRS failed to the maximum value of it's 100 degree range (615°F).

C. Incorrect: On a plant trip the Quick Open (QO) signal from the RRS will clear (reset). Tave will lower to ~ 540°F on the plant trip which is below 554 °F temperature needed for a QO and above the 535°F temperature where the Area Demand signal will close the CDVs.
Plausible: The failed high RCS Hot Leg temperature will keep Tave > 554°F longer therefore maintain the CDVs full open longer. But with the other three good temperature indications Tave will lower to less than 554°F and remove the QO signal. The examinee must understand how Tave is calculated, what normal RCS T_c and T_h values are, and when the RRS system throws out failed indicators.

D. Incorrect: Letdown flow will not go to the maximum allowed by the Letdown Limiter because this failure will not change the programmed level setpoint.
Plausible: Letdown flow would go to the maximum allowed by the Letdown Limiter if a Hot Leg temperature input failed to its minimum value of its 100 degree range (515°F).

References: RRS-01-C Rev 5/3 Student Ref: NONE

Learning Objective: 282175 (03170-MB) Describe the response of the Reactor Regulating System to a failed RCS temperature or steam pressure input, during steady state and non-steady state conditions.

Question Source: Bank

Question History: 413487 (71180-MB) no knowledge of use on an NRC exam

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: CFR 41.7 / 45.7

Comments (Question 61):
<<

>>QUESTION 62

The plant is at 100% power.

- Main Generator MWe output has decreased.
- RCS Cold Leg temperature (T_{COLD}) is 545 °F and stable.

What will cause the above indications?

A. An Atmospheric Dump Valve (ADV) partially opens.

B. A malfunction of the Steam Jet Air Ejector (SJAE).

C. Lowering Circulating Pump speeds from 100 to 90% (T_{INTAKE} = 45 °F).

D. An Inadvertent SIAS occurred causing boration of the RCS.

<<

>>QUESTION 62

K&A Rating: 055K3.01 (2.5)

K&A Statement: Condenser Air Removal System (CARS). Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser.

Key Answer: **B**

Justification:

A. Incorrect: An Atmospheric Dump Valve (ADV) partially opening will lower Mwe but will also lower RCS Cold Leg temperature.
Plausible: A partially open ADV will lower Mwe.

B. Correct: If the Steam Jet Air Ejector (SJAE) malfunctions it will no longer function to remove non-condensable gases from the Main Condenser. As non-condensable gases build up around the condenser tubes heat transfer lowers and Condenser temperature rises. Since the Condenser is in a saturated state the enthalpy of the water in the Hotwell will rise. This rise in enthalpy lowers the delta enthalpy in the system, resulting in Main Generator output lowering.

C. Incorrect: Lowering Circulating System flow with Intake water temperature of 45 °F will not lower MWe. It may actually raise Mwe by lessening condensate depression (subcooling hotwell water). And will not lower RCS Cold Leg temperature.
Plausible: Lowering Circulating Water system flow when Intake temperature is much warmer, when there is no condensate depression will degrade condenser vacuum and lower Mwe without changing RCS Cold Leg temperature.

D. Incorrect: An Inadvertent SIAS occurring causing boration of the RCS, will lower MWe, but will also lower RCS Cold Leg temperature.
Plausible: An Inadvertent SIAS occurring causing boration of the RCS will lower MWe.

References: CAR-01-C Rev 5, GFES Thermodynamic Cycles Chapter 5 Rev 4
Student Ref: None

Learning Objective: 280968 (MB-02214) Describe the effects of a loss or malfunction of the Condenser Air Removal System on the Main Condenser.

Question Source: New

Question History: New. Question concept from McGuire Plant NRC 2007 RO Q35

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7 / 45.6

Comments (Question 62):

02SEP20 DF – Changed distractors C & D based on Don Jackson review. C changed winter to T_{INTAKE} of 45 degrees. D changed ppm difference in CH pumps to boration of RCS. John updated justification for Don Jackson's recommended changes on 09/09/2020.

08/07/2020 jwr. Made recommended NRC change from “could” in question stem to “will”.

6/8/20 jwr – changed answer “B” from the SJAE steam supply closes” to a malfunction of the SJAE. This was done because the of the low success rate on this question and that Joe Browning felt that temperature would not remain stable (it might not change enough to see it but it has to change because steam flow lowered when the valve closes.) if main steam is isolated to the SJAE. <<

>>QUESTION 63

The “A” Waste Gas Decay Tank (WGDT) discharge is in progress. The Waste Gas Radiation Monitor, RM-9095 alarms as a result of high radioactivity levels.

What are the valves that automatically CLOSE to stop the tank discharge?

A. “A” WGDT inlet 2-GR-6.1A and Waste Gas Discharge Isolations 2-GR-37.1 & 2-GR-37.2.

B. “A” WGDT inlet 2-GR-6.1A and Pressure Control valve 2-GR-9.1.

C. “A” WGDT outlet 2-GR-8.1A and Waste Gas Discharge Isolations 2-GR-37.1 & 2-GR-37.2.

D. “A” WGDT outlet 2-GR-8.1A and Pressure Control valve 2-GR-9.1.

<<

>> QUESTION 63

K&A Rating: 071K4.04 (2.9)

K&A Statement: Waste Gas Disposal System (WGDS). Knowledge of design feature(s) and/or interlock(s) which provide the following: Isolation of waste gas release tanks.

Key Answer: **C**

Justification:

A. Incorrect: A high radiation condition on the Waste Gas Radiation Monitor, RM-9095 does not close the "A" WGDT inlet 2-GR-6.1A.
Plausible: A high radiation condition on the Waste Gas Radiation Monitor, RM-9095 does automatically closethe Waste Gas Discharge Isolations 2-GR-37.1 & 2-GR-37.2. Closing 2-GR-6.1A would isolate the tank from the system and closing 2-GR-37.1 and GR-37.2 would stop the discharge.

B. Incorrect: Neither the "A" WGDT inlet 2-GR-6.1A nor the Pressure Control valve 2-GR-9.1 automatically close from a high radiation signal on Waste Gas Radiation Monitor, RM-9095.
Plausible: Closing these two valves would isolate the discharge and isolate the tank from the system.

C. CORRECT: The "A" Waste Gas Decay Tank (WGDT) outlet 2-GR-8.1A and Waste Gas Discharge isolations 2-GR-37.1 & 2-GR-37.2 automatically close from a high radiation signal on Waste Gas Radiation Monitor, RM-9095 to stop the discharge.

D. Incorrect: A high radiation signal on the Waste Gas Radiation Monitor, RM-9095 does not close the Pressure Control valve 2-GR-9.1.
Plausible: A high radiation condition on the Waste Gas Radiation Monitor, RM-9095 does automatically close the "A" WGDT outlet 2-GR-8.1A and closing the Pressure Control valve 2-GR-9.1 would also isolate the discharge.

References: GRW-04-C Rev 7/1, ARP 2590H-016 Rev 000 Student Ref: NONE

Learning Objective: 287623 (00513-MB) State the interlocks and trips associated with the following system component: (A) Waste gas compressors, (B) WGDT inlet and outlet valves, (C) Waste gas radiation monitor

Question Source: Bank

Question History: 418940 (0053701-MB). No record of use on a NRC exam. The stem and answers were changed slightly for clarity, and the justification was written.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.7

Comments (Question 63): 6/8/20 jwr – changed answers to remove the waste gas discharge isolations 2-GR-37.1 and 2-GR-37.2 as an answer since this is what the majority of the validators were selecting. This answer is correct for liquid waste discharges. Made the question into a 2X2. OPS 5/19/20 jwr – changed question based on first OPS validation comments. Stem did not say

ALL valves that automatically close which resulted in multiple correct answers. Changed answer “D” since the tank inlet would be closed if it was being discharged, thereby making answer more plausible.<<

>>QUESTION 64

The plant is operating at 100% power. Containment venting (depressurization) is in progress.

1. What flow path is being used? AND
2. What radiation monitor will isolate the flow path if a high radiation condition in Containment occurs?

A. 1. Hydrogen Purge valves EB-99 & 100 or EB-91 & 92.
   2. Containment High Range Radiation Monitors RM-8240 or RM-8241.

B. 1. Hydrogen Purge valves EB-99 & 100 or EB-91 & 92.
   2. Containment Radiation Monitors RM-8123A/B or RM-8262A/B.

C. 1. Containment Purge Isolation Dampers AC-5, 6, 7 & 8.
   2. Containment High Range Radiation Monitors RM-8240 or RM-8241.

D. 1. Containment Purge Isolation Dampers AC-5, 6, 7 & 8.
   2. Containment Radiation Monitors RM-8123A/B or RM-8262A/B.

<<

>>QUESTION 64

K&A Rating: 029A1.02 (3.4)

K&A Statement: Containment Purge System (CPS). Ability to predict and/or monitor changes in parameters to prevent exceeding design limits associated with operating the Containment Purge System controls: Radiation levels.

Key Answer: **A**

Justification:

A **CORRECT:** The Hydrogen Purge valves (EB-99 & 100 or EB-91 & 92) are used to vent Containment and the Containment High Range radiation monitors RM-8240 & RM-8241 will automatically close the Hydrogen Purge valves at 5 R/hr. In MODES 1-4 the Containment Purge Isolation Dampers AC-5, 6, 7 & 8 are required to be closed and prevented from opening.

B. **Incorrect:** The Containment Radiation Monitors RM-8123A/B or RM-8262A/B will automatically close the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8) not the Hydrogen Purge valves (EB-99 & 100 or EB-91 & 92).
Plausible: Containment is purged in MODES 5 & 6 using the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8) and the Containment Radiation Monitors RM-8123A/B or RM-8262A/B will automatically close the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8).

C. **Incorrect:** The flow path is not through the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8) in MODES 1-4.
Plausible: Containment is purged in MODES 5 & 6 using the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8)

D. **Incorrect**: The flow path is not through the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8) in MODES 1-4.
Plausible: Containment is purged in MODES 5 & 6 using the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8) and the Containment Radiation Monitors RM-8123A/B or RM-8262A/B will automatically close the Containment Purge Isolation Dampers (AC-5, 6, 7 & 8).

References: RMS-00-C Rev 7/8, OP 2314B Rev 026-00, ARP 2590A-119 Rev 000, Technical Specification Rev through change No. 297
Student Ref: NONE

Learning Objective: 282105 (03129-MB) Describe the automatic actions and logic associated with the Containment High Range Radiation Monitoring Subsystem.
Question Source: New
Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.5 / 45.5

Comments (Question 64): 08/07/2020 jwr. Completed NRC's recommended change from 7/24/20 to put question marks in stem questions.<<

>>QUESTION 65

The plant is at 80% power with Bus 24E aligned to Bus 24C.

A loss of bus 22E occurs and the crew has entered AOP 2503E, “Loss of Vital 480 VAC Bus 22E”.

1. How will the Facility 1 Service Water pump be aligned by AOP 2503E? AND

2. How will the Service Water Strainer be aligned?

A. 1. "B" Service Water pump operating powered from Unit 3.
   2. "B" Service Water Strainer is powered from Facility 2.

B. 1. "B" Service Water pump operating powered from Facility 1.
   2. "B" Service Water Strainer is powered from Facility 2.

C. 1. "B" Service Water pump operating powered from Unit 3.
   2. "B" Service Water Strainer is powered from Facility 1.

D. 1. "A" Service Water pump operating powered from Facility 1.
   2. "A" Service Water Strainer is powered from Facility 1.

<<

>>QUESTION 65

K&A Rating: 075K2.03 (2.6)

K&A Statement: Circulating Water System. Knowledge of bus power supplies to the following: Emergency/essential SWS pumps.

Key Answer: **A**

Justification:

A. **CORRECT:** “B” Service Water (SW) pump powered from Unit 3 and “B Service Water strainer (SWS) powered from Facility 2 is correct. This ensures two SW pumps are not powered from the same EDG and will allow operation of the “B” SWS. Both tie breakers, A305 24C/24E and A408 24D/24E must be open to allow Facility 2 power to be aligned to the “B” SWS.

B. **Incorrect:** "B" SW pump operating powered from Facility 1 is not correct. Since the Facility 1 strainer power is not available for the “B” strainer.
Plausible: The “B” SW pump can be aligned to either Facility 1 or 2 and power is still available from Facility 1, i.e 4.16 KV bus 24C has not been lost.

C. **Incorrect**: "B" Service Water Strainer powered from Facility 1 is not correct. Bus 22E provides power to the “B” SWS and the strainer therefore has no power.
Plausible: The “B” Service Water (SW) pump powered from Unit 3 is correct and the the “B” SWS can be aligned to either Facility 1 or 2. The examinee may not remember where the “B” SWS can be powered from. The “B” SWS Facility 1 power is from MCC B51 that is powered from Bus 22E..

D. **Incorrect**: “A” SW pump operating powered from Facility 1 is not correct.
Plausible: The “A” SW pump still has power available from it’s normal supply bus 24C and the examinee could reason that there is another way to power the “A” SWS from Facility 1.

References: AOP 2503E Rev 004-00 A03-01-C Rev 1/2 Student Ref: NONE

Learning Objective: 282827 (05634-MB) Outline the major actions for AOP 2503A/B/C/D/E/F, Loss of 480 VAC Bus 22A, 22B, 22C, 22D, 22E, or 22F.

Question Source: Bank

Question History: 64596-MB. There is no record of use on an NRC exam. The question stem and answers were changed to make question and answers easier to read. The justification was enhanced to better explain answers and why incorrect answers are plausible.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7

Comments (Question 65): 08/08/2020 jwr. Completed change requested by the NRC on 7/24/20. They requested that answer “C” part 1 “Facility 2” be changed to “Unit 3”. This would make two answers with the correct “powered from Unit 3” and this raises the level of difficulty.

5/18/20 jwr changed stem, minor editorial, and changed answers to spell out service water pump and service water strainer because SWS is not an acronym operators are familiar with. OPS validation comment.
<<

>>QUESTION 66

A Reactor Operator (RO) has stood the following watches:

- (3) 12 hour watches in the Balance of Plant (BOP) position in May.
- (1) 12 hour watch in the BOP position in June.

Per OP-AA-103, “Operator Qualifications”, in order to maintain an active status in the At-The-Controls (ATC) position,

The RO ___1.___ and the latest they can stand a watch is ___2.___?

	1.	2.
A.	must stand a watch in the ATC position	June 30
B.	can stand a watch in either the ATC or BOP position	June 30
C.	must stand a watch in the ATC position	July 31
D.	can stand a watch in either the ATC or BOP position	July 31

<<

>> QUESTION 66

K&A Rating: G2.1.4 (3.3)

K&A Statement: Conduct of Operations. Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, “no-solo” operation, maintenance of active license status, 10CFR55, etc.

Key Answer: **B**

Justification:

A. **Incorrect:** Must stand the watch in the ATC position is not correct. The requirement is to stand a minimum of five 12-hour shifts per calendar quarter in the applicable Technical Specification (TS) licensed position. The applicable TS licensed position is Reactor Operator (RO) and this position is both ATC and BOP.
Plausible: Examinee could could think that at least one watch must be stood in each position to maintain active status. This is similar to the requirements for a Senior Reactor Operator (SRO). Licenses performing both SRO and RO duties must stand one 12 hour watch as a SRO to maintain that portion of their license active.

B. **CORRECT:** Can stand the watch in either the ATC or BOP position and the latest it can be stood is June 30 is correct. The requirement is to stand a minimum of five 12-hour shifts per calendar quarter in the applicable Technical Specification (TS) licensed position. The applicable TS licensed position is Reactor Operator (RO) and this position is both ATC and BOP, and the calendar quarter runs from April through June.

C. **Incorrect**: Must stand the watch in the ATC position and the latest they can stand the watch is July 31 is not correct. The watch can be stood in either the ATC or BOP and July 31 is outside the calendar quarter (April, May, and June).
Plausible: Examinee could could think that at least one watch must be stood in each position to maintain active status and that the watches need to be stood in a quarter not necessarily a calendar quarter.

D. **Incorrect**: The latest they can stand the watch is July 31 is not correct.. July 31 is outside the calendar quarter (April, May, and June).
Plausible: Examinee could could think that watches need to be stood in a quarter not necessarily a calendar quarter.

References: OP-AA-103 rev 7 Student Ref: NONE

Learning Objective: 251451 As outlined in OP-AA-103, Operator Qualifications, describe the licensed Operator responsibilities for maintaining qualifications.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.10

Comments (Question 66):

09/02/2020 jwr. Don Jackson identified a spelling error in stem. Position was mis-spelled and was fixed by dmf.

20AUG20 df Changed ‘must’ to ‘can’ in answers B & C to get rid of possible subset error per NRC comments.

08/12/20 jwr. Completed changes requested by the NRC identified on 7/24/20. The NRC identifies that the question was being asked about maintaining a SRO license active. Their comment was this is SRO level. Wrote a new question on maintaining active status for a RO.

6/8/20 jwr – changed to have answer ‘BOP and RO only” to be more in line with an RO question. Also made the correct answer “B” to equalize answers between letters.

<<

>>QUESTION 67

Which of the following statements correctly describes the Technical Specification requirement for staffing in MODES 1 – 6?

A. A Shift Technical Advisor is required on-shift.

B. A Health Physics Technician is required on-site.

C. Two Plant Equipment Operators are required on shift.

D. A Chemistry Technician is required on-site.

<<

>>QUESTION 67

K&A Rating: G2.1.5 (2.9)

K&A Statement: Conduct of Operations. Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Key Answer: **B**

Justification:

A. **Incorrect:** A Shift Technical Advisor is only required in MODES 1-4.
Plausible: Millstone 2 staffs the STA position in all MODES.

B. **Correct:** A Health Physics Technician is required on site when fuel is in the reactor. This is contained in the specification in the note under the Minimum Shift-Crew Composition. All MODES (1-6) have fuel in the reactor.

C. **Incorrect**: In MODES 5 and 6 only one Plant Equipment Operator (Non-Licensed Operator) is required.
Plausible: Normally there is one additional PEO that staffs the Appendix R required position. But this is not a requirement of the Technical Specification.

D. **Incorrect:** A Chemistry Technician is not required by Technical Specifications to be on-site in MODES 1-6.
Plausible: A Chemistry Technician is part of E-Plan and rotates shifts like Operations and Health Physics. And attend all start of shift briefs.

References: Technical Specifications section 6.2 Rev. though change No. 397,
OP-AA-100 Rev 40 Student Ref: NONE

Learning Objective: 282397 (02405-MB) Given an Operational Mode and a staffing condition, evaluate the adequacy of the crew composition as given in T/S or the TRM. This is an Objective taught to the Reactor Operators and they are responsible for.

Question Source: Bank

Question History: 413757 (54481-MB) No indication it has been used on a previous NRC exam. One answer was changed to add plausibility. The stem and answers were changed slightly to shorten.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.10

Comments (Question 67): 08/08/2020 jwr. Completed changes to address the NRC's comments on 7/24/20. The comments were to remove the first sentence in the stem, to add "staffing in" after

“requirement for” in the second sentence of stem, and to put in to the justification that this is expected RO knowledge. There was some discussion whether this was RO knowledge and in the end NRC agreed with using the question since it validated OK and with making the change recommended.

5/19/20 jwr – First OPS validation comment was to change “all MODES” to “MODES 1-6” to ensure no confusion on MODE 0 where an HP technician is not required. Also replaced answer “D” with a Chemistry Technician to balance answers with Operations answers and other department answers.
<<

>>QUESTION 68

A Technical Specification required Quarterly (92 days) surveillance was last performed 95 days ago.

1. What is status of the equipment?

2. The equipment will remain in its current status …

A. 1. OPERABLE.
   2. as long as the surveillance is completed at anytime within the next three months.

B. 1. OPERABLE.
   2. as long as surveillance is completed within an additional 25% of the 92 days.

C. 1. NOT OPERABLE.
   2. until the missed surveillance and a risk evaluation are performed within 24 hours.

D. 1. NOT OPERABLE.
   2. until the missed surveillance is completed.

<<

>>QUESTION 68

K&A Rating: G2.2.37 (3.6)

K&A Statement: Equipment Control. Ability to determine operability and/or availability of safety related equipment.
K/A match: ROs are trained on surveillance intervals and associated Technical Specification 4.0.2 and 4.0.3 under the learning objective listed.

Key Answer: **B**

Justification:

A. **Incorrect:** LCO 4.0.3 does not yet require action under the equipment LCO because there is still 115-92 = 23 days left to perform the surveillance under the 25% extension allowed by LCO 4.0.2.
Plausible: Applicant may confuse the use of LCO 4.0.2 with LCO 4.0.3.

B. **CORRECT**: Operable in accordance with LCO 4.0.2. A maximum extension, not to exceed 25%, of the surveillance time interval (25% of 92 days is 115 days) is allowed.

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.
Plausible: LCO 4.0.3 states that if it is discovered that Surveillance was not performed within its specified surveillance interval, then compliance with the requirement to declare the Limiting Condition for Operation not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater. The applicant may only remember the 24 hour extension.

D. **Incorrect**: The equipment is still Operable. 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.
Plausible: Applicant may believe that failing to meet the surveillance frequency requirement deems the equipment inoperable. This would be the case if the equipment failed the surveillance.

References: Technical Specification 4.0.2 and 4.0.3 Rev through change No. 397
Student Ref: NONE

Learning Objective: 282396 (01864-MB) As given in Technical Specifications, state the maximum allowed extension of a surveillance interval and the principles governing use of an extension.

Question Source: Bank
Question History: 413754 (0086709-MB) 2014 NRC exam Q97, significant changes to question; stem, answers, and justification but wouldn't characterize as a modified question.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.7 / 43.5 / 45.12

Comments (Question 68): 08/08/20 jwr. Completed changes recommended by the NRC on 7/24/20. Their recommendation was to add “at anytime” after “completed” in answer A(2) and put in the justification that ROs are trained to this level.
<<

>>QUESTION 69

The plant was operating at 100% power, with all CEAs fully withdrawn, when Regulating Group 7 CEA #1 dropped to the bottom of the core.

The crew lowered reactor power to less than 70% in one hour.

After one hour and 52 minutes since the CEA dropped, I&C informs the control room that the CEA recovery steps may begin.

What action is required per AOP 2556, CEA Malfunctions?

A. Do NOT perform CEA recovery steps. Immediately commence a plant shutdown to MODE 3.

B. Ensure the dropped CEA is withdrawn to at least 170 steps within the next 20 minutes.

C. Enter LCO 3.0.3 and withdraw the CEA to at least 170 steps within the next 60 minutes.

D. Trip the plant and maintain the reactor shut down for a minimum of 2 hours before restarting.

<<

>>QUESTION 69

K&A Rating: G2.2.40 (3.4)

K&A Statement: Equipment Control. Ability to apply Technical Specifications for a system

Key Answer: **A**

Justification:

A. **Correct:** TSAS 3.1.3.1 action A.1 and AOP 2556 require reduction of thermal power to < 70% within one hour and the restoration of the misaligned CEA within 10 steps of all other CEAs in its' group within 2 hours, or otherwise be in MODE 3 within the next 6 hours.

B. **Incorrect:** Taking 20 minutes to withdraw the CEA would exceed the 2 hour TS requirement. There is only 8 minutes left till the 2 hour limit.
Plausible: The candidate may misapply the TS and believe they have 2 hours after reaching 70% to withdraw the CEA. Also the examinee may not know the TS or procedure requirements.

C. **Incorrect**: TS 3.03 does not need to be entered because while the TS 3.1.3.1 LCO is not met the associated action statement can be met. TS 3.0.3 is used when a TS does not provide specific direction for the condition. This TS has direction for this condition.
Plausible: The examinee may not know the TS or procedure requirements.

D. **Incorrect:** The procedure, nor the TS require a plant trip. A plant trip is not required and would be considered an overly aggressive plant shutdown and a non-conservative action.
Plausible: The examinee could reason that since the CEA could not be realigned within two hours that they are outside the design bases and the plant should be tripped. They may not fully understand the TS or procedure requirements.

References: TS 3.1.3.1 Rev through change No. 397, AOP 2556 Rev 022 Student Ref: NONE

Learning Objective: 281014 (02251-MB) Given a list of plant conditions or parameter values, a copy of Technical Specifications and an operational situation affecting CEDS, identify the applicable action statement requirements in accordance with plants Technical Specification and plant operating procedures.

Question Source: Bank

Question History: 453217 (0080625-MB) 2008 NRC exam Q71, made editorial stem changes to shorten stem, modified justification to strengthen and add plausibly statements.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 41.10

Comments (Question 69):08/08/2020 jwr. Completed changes to incorporate the NRC comments received on 7/24/20. The NRC recommended that the stem be changed to state …What action is

required per the governing procedure…AOP 2556, CEA Malfunctions. They also recommended that answer “B” time be changed from 10 minutes to 20 minutes.

5/18/20 jwr minor change from 1 hour and 50 minutes to one hour and 52 minutes. OPS validation comment.
<<

>>QUESTION 70

The BOP is performing a task preview for the “A” Service Water pump surveillance. The pump delta-pressure (DP) acceptance criteria is identified as incorrect by Engineering and a change is required.

In accordance with AD-AA-100, “Technical Procedure Process Control”,

1. What process will be used to make the change? AND
2. What is required to implement the change?

A. 1. Administrative Correction Process.
   2. Handwrite change in procedure and have supervisor initial and date.

B. 1. Administrative Correction Process.
   2. Complete Administrative Correction Process form and obtain approvals.

C. 1. Feedback Incorporation Process (FIP).
   2. Handwrite change in procedure and have supervisor initial and date.

D. 1. Feedback Incorporation Process (FIP).
   2. Complete FIP form and obtain approvals.

<<

>>QUESTION 70

K&A Rating: G2.2.6 (3.0)

K&A Statement: Equipment Control. Knowledge of the process for making changes to procedures.

Key Answer: **D**

Justification:

A. **Incorrect:** The Administrative Correction Process can't be used for quantative acceptance criteria. This is detailed in AD-AA-100 under section 5.3.5, Administrative Correction.
Plausible: The Administrative Correction Process can be used for various types of changes and the normal method for making an administrative correction is to handwritte the change in the procedure and have it initialed and dated by a supervisor.

B. **Incorrect:** The Administrative Correction Process can't be used for quantative acceptance criteria. This is detailed in AD-AA-100 under section 5.3.5, Administrative Correction.
Plausible: The Administrative Correction Process can be used for various types of changes and there is an Administrtaive Correction Authorization form.

C. **Incorrect**: Use of the FIP process requires completion of the FIP form including obtaining approvals.
Plausible: Some procedure changes only required the change to be handwritten in the procedure and initialed and dated by a supervisor.

D. **CORRECT:** The Feedback Incorporation Process (FIP) and complete FIP form and obtain approvals is correct. AD-AA-100 section 3.1 has a note that states that the Feedback Incorporation Process (FIP) is preferred method to initiate new or revised single-site procedures.

References: AD-AA-100 Rev 12 Student Ref: NONE

Learning Objective: 251837 Describe how a procedure change may be made when needed immediately.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:
10CFR55: CFR 41.10 / 43.3 / 45.13

Comments (Question 70): 6/21/20 jwr – John W comment to say identified incorrect by Engineering. Change to stem made. 6/9/20 jwr. This was a high missed question. Revised question to specify what is normally used by Operators. 5/21/20 jwr. This question was replaced. The initial OPS validation commented that this question is not something the examinees are expected to know without the procedure, i.e. what is an intent vs non-intent change and what approval level is required. 6/10/20 added "by Engineering" in stem.

<<

>>QUESTION 71

A Plant Equipment Operator is performing a valve lineup and must enter an area where the highest dose rate is 1500 mR/hr.

Which one of the following describes the radiological requirements for posting and control of this area?

This area is required to be posted as a __1__, AND

Personnel entering the area are required to maintain each entrance closed and properly secured except __2__.

A. 1. High Radiation Area.
   2. When inside the area so personnel are not prevented from leaving the area.

B. 1. Locked High Radiation Area.
   2. When inside the area so personnel are not prevented from leaving the area.

C. 1. Locked High Radiation Area.
   2. For periods of ingress or egress, unless guarded to prevent unauthorized entry.

D. 1. High Radiation Area.
   2. For periods of ingress or egress, unless guarded to prevent unauthorized entry.

<<

>>QUESTION 71

K&A Rating: G2.3.12 (3.2)

K&A Statement: Radiation Control: Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Key Answer: **C**

Justification (Question 71):

A. **Incorrect:** The area radiation reading >1000 mR/hr requires it to be posted as a locked high radiation area. The entrance to each locked high radiation area must remain closed and locked except for periods of ingress and egress unless the entrance is guarded to prevent unauthorized entry.
Plausible: This is a plausible answer for a candidate that incorrectly classifies the radiation area. An applicant may misinterpret the requirement to not restrict egress from the area as the ability to maintain the opening unlocked.

B. **Incorrect:** The area radiation reading >1000 mR/hr requires posting this area as a locked high radiation area is correct. The entrance to each locked high radiation area must remain closed and locked except for periods of ingress and egress unless the entrance is guarded to prevent unauthorized entry so the second part is incorrect.
Plausible: An applicant may misinterpret the requirement to not restrict egress from the area as the ability to maintain the opening unlocked.

C. **CORRECT:** The area radiation reading >1000 mR/hr requires posting this area as a locked high radiation area. The entrance to each locked high radiation area must remain closed and locked except for periods of ingress and egress unless the entrance is guarded to prevent unauthorized entry.

D. **Incorrect:** The area radiation reading >1000 mR/hr requires it to be posted as a locked high radiation area.
Plausible: The controls in Item 2 are correct in this answer. This is a plausible answer for a candidate that incorrectly classifies the radiation area.

References: Student Ref: NONE

Learning Objective: 25195 List the Definition and entry requirements for “Rad Area”, “High Radiation Area (HRA)”, “Locked High Radiation Area (LHRA)”, “Very High Radiation Area (VHRA)”.

Question Source: Bank

Question History: 451934 (2014039-MB) 2014 NRC Exam Q71. Stem changed to shorten length and made editorial changes to justification. The 2014 NRC was written by the NRC.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.12

Comments (Question 71):

02SEP20 DF – changed ‘locked’ to ‘properly secured’ based on Don Jackson comments.

08/08/2020 jwr. Completed changes recommended by the NRC on 7/24/20. The NRC recommended that the stem be changed for doing an operator task, such as a operating a valve. The stem was modified to state a Plant Equipment Operator is performing a valve lineup.

>>QUESTION 72

The Charging Pump Area Radiation Monitor (RM-7894) alarms.

The US directs the RM-7894 module on RC-14 be placed in ALARM DEFEAT.

Why is the radiation monitor's switch placed in the ALARM DEFEAT position?

A. To silence the radiation monitor's horn on the local module.

B. To reset any automatic action caused by the radiation monitor.

C. To clear the radiation monitor's red and/or amber lights on RC-14.

D. To allow other radiation monitor alarms to annunciate on C-06/7.

<<

>>QUESTION 72

K&A Rating: G2.3.15 (2.9, 3.1)

K&A Statement: Radiation Control: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Key Answer: **D**

Justification:

A. **Incorrect:** The applicable ALARM DEFEAT switch does not silence the local alarm. The alarm is silenced by turning the local key-switch to "OFF". In fact if the switch is placed in ALARM DEFEAT with no alarms present, it will cause the local panel's audible alarm to sound.
Plausible: It would be desired to silence the local alarm. It is reasonable the examinee think that going to ALARM DEFEAT would defeat the functions of the radiation monitor (RM) and silence the local audible alarm.

B. **Incorrect:** The ALARM DEFEAT switch will NOT reset any automatic action caused by the RM. In fact, the ALARM DEFEAT switch will result in a RM failure which will prevent resetting any automatic function.
Plausible: It is reasonable the examinee believe that the ALARM DEFEAT switch would clear any RM alarm(s) and therefore reset any automatic action.

C. **Incorrect:** The ALARM DEFEAT switch will NOT clear the red and amber lights on the RC-14 module. In fact, the ALARM DEFEAT switch will cause the red and amber lights to be lit.
Plausible: It is reasonable the examinee believe that the ALARM DEFEAT switch would clear the RED HI RAD ALARM light when the RM is taken to ALARM DEFEAT.

D. **CORRECT:** Placing the applicable ALARM DEFEAT switch in the ALARM DEFEAT position will allow other area radiation monitor alarms to be annunciated on C-06/7. That is the purpose of the switch.

References: RMS-00-C R7C8, ARP 2590E-128 Rev 000-01 Student Ref: NONE

Learning Objective: 282113 (06435-MB) Describe the function of the RM controls on RC-14 and how the controlled components(s) is/are affected by each mode or position.

Question Source: Bank

Question History: 1000109-MB 2008 NRC exam Q73. Changed stem and significantly enhances question justification.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.12 / 43.4 /45.9

Comments (Question 72):
<<

>>QUESTION 73

Which one of the following correctly describes the proper use of Event Specific EOPs?

A. Steps identified as TCOA (Time Critical Operator Action) may be performed in any sequence as determined by the US/SM.

B. Steps marked with an asterisk (*) may be brought forward by the SM/US to restore or preserve a safety function.

C. Actions which deviate from the written sequence of steps of EOPs may be performed when appropriate guidance for that action is found in an AOP.

D. Steps marked with an asterisk (*) sign may be performed in any sequence without US/SM direction.

<<

>>QUESTION 73

K&A Rating: G2.4.19 (3.4)

K&A Statement: Emergency Procedures/Plan: Knowledge of EOP layout, symbols, and icons.

Key Answer: **B**

Justification:

A. **Incorrect:** This is incorrect because only steps marked with an asterisk (*) may be brought forward by the SM/US to restore or preserve a safety function. Steps marked as a TCOA must be completed within a specified time to ensure that the plant complies with the regulatory commitments and assumptions related to the safety analyses and other licensing basis events. There is no specific guidance they can be performed in any sequence determined by the US/SM.
Plausible: Completion of TCOAs are critically important to plant safety. The examinee could reasonable believe that the US/SM could implement them out of sequence.

B. **CORRECT**: Steps marked with an asterisk (*) may be brought forward by the SM/US to restore or preserve a safety function.

C. **Incorrect:** EOP sequencing must be followed unless the step is asterisked and brought forward by the SM/US to restore or preserve a safety function.
Plausible: AOPs can be used with an EOP but the EOP is always the controlling document.

D. **Incorrect**: Steps marked with an asterisk (*) sign may only be performed in any sequence without US/SM direction is not correct. US/ SM direction is required.
Plausible: Examinee may believe the asterisk (*) allows steps to be performed out of sequence without US direction as long as the actions do not interfere with maintaining an existing safety function. This is acceptable if directed by the US.

References: OP 2260 Rev 015-00, EOP 2532 Rev 035-00 Student Ref: NONE

Learning Objective: 283638 (MB-05270) Outline the EOP usage rules as described in OP 2260, Unit 2 EOP User's Guide.

Question Source: Bank

Question History: 453551 (0054207-MB) NRC 2005 ILT exam Q75. No change to stem, changed answer "A" from Plus (+) sign to TCOA, and editorial changes to answers "B" and "C" to make them more in line with OP 2260 guidance.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.10 / 45.13

Comments (Question 73):
02SEP20 DF – changed from COG level Comprehensive/Analysis to Memory based on Don Jackson review comments

08/08/2020 jwr. Completed changed recommended by the NRC on 7/24/20. The NRC recommended that the justification for answer “A” be enhanced to specify why answer “A” is not correct. The justification was changed to clearing state only asterisked step can be pulled forward. 6/23/20 jwr – changed answer “D” slightly since the US can direct steps to be performed out of sequence, if the actions do not interfere with maintaining an existing safety function.
<<

>>QUESTION 74

The Main Control Boards has panel labels to identify components and instruments.

Which label is used to identify post-accident monitor (Reg Guide 1.97) equipment?

A. HSD.

B. Red Dot.

C. Black Triangle.

D. Black Dot.

<<

>>QUESTION 74

K&A Rating: G2.4.3 (3.7, 3.9)

K&A Statement: Ability to identify post-accident instrumentation.

Key Answer: **C**

Justification:

A. **Incorrect:** The "HSD" label on the Main Control Boards is powder blue with white letters and indicates control or indicator has a duplicate component on the Hot Shutdown Panel (C21), or the Appendix "R" Panel (C10).
Plausible: HSB is a Main Control Board label.

B. **Incorrect**: The "Red Dot" label on the Main Control Boards indicates Z1 powered equipment.
Plausible: The "Red Dot" is a Main Control Board label.

C. **CORRECT:** The "Black Triangle" label on the Main Control Boards indicates Reg. Guide 1.97 (accident monitoring) equipment.

D. **Incorrect**: The "Black Dot" label on the Main Control Boards indicates Z5 powered swing equipment.
Plausible: The "Black Dot" is a Main Control Board label.

References: 25203-30210 revision 10, SP-EE-261 revision 4. Student Ref: NONE

Learning Objective:

Question Source: New

Question History: Written from similar questions in the Industry Bank; like Q74 of Turkey Point 2011 SRO NRC exam.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: CFR 41.6 / 45.4

Comments (Question 74): 6/21/20 jwr. John W comment stem should say "post" accident monitor equipment. Change made. 5/21/20 jwr. This question was replaced with a new question that specifically addressed the K/A. The initial OPS validation was that this question is really bug dust and something that is not operationally focused. The new question, we believe addresses their comments.

<<

>>QUESTION 75

The plant is operating at 87% power. "A" and "B" condensate pumps are in service.

- Alarm CONDENSATE PUMP A DIS PRES LO (C-05, B-11) annunciates.
- CND HDR PRES, PI-5224 is reading 415 psi.

What action is taken for this condition?

A. Start the "C" condensate pump.

B. Manually raise main feed pump speed.

C. Trip the reactor.

D. Manually lower main feed pump speed.

<<

>>QUESTION 75

K&A Rating: G 2.4.50 (4.0)

K&A Statement: G 2.4.50 Generic, Emergency Procedures /Plan: Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.

Key Answer: **A**

Justification:

A. CORRECT: Condensate Header pressure of 400 psi causes CONDENSATE PUMP B DIS PRES LO (C-05, C-11) to annunciate. The ARP for this annunciator (ARP 2590D-43) directs the operator start additional condensate pumps if condensate header discharge pressure is less than 460 psig.

B. Incorrect: Raising the speed on a SGFP will cause condensate header pressure to lower
Plausible: The examinee thinks that raising SGFP speed raises suction pressure.

C. Incorrect: 400 psig condensate header pressure equates to 260 psig SGFP suction pressure SGFP suction trip is 245 psig for greater than 30 seconds.
Plausible: The examinee thinks a SGFP tripped.

D. Incorrect: Lowering SGFP speed raises suction pressure temporarily until the FRVs react.
Plausible: The examinee doesn't think about FRV response.

SRO Justification:

References: ARP 2590D-43
Student Ref: NONE

Learning Objective: Describe the effects of a loss or malfunction of the Condensate system on the following: A) Feedwater System B) Feedwater Heater Drains and Vents System.

Question Source: Bank

Question History: 2002 NRC

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 75): Changed stem to include alarms, changed syntax to singular action in response to NRC comments. DF 03AUG20
<<

>>QUESTION 76

The plant is at 100% when the Turbine trips. The following conditions exist:

- The Turbine trip successfully tripped the Reactor.
- A Power Operated Relief Valve (PORV) opened on the trip and has not closed.
- A loss of Offsite Power occurred on the trip.
- Subcooling is less than 30 °F and lowering.
- No secondary radiation monitors are in alarm or rising.
- One Pressurizer level indication is reading 10% and the other is reading 80%.
- EOP 2525, Standard Post Trip Actions has been completed.

1. What EOP will the Unit Supervisor direct be entered? AND

2. Which Pressurizer level is consistent with the current condition?

A. EOP 2532, Loss of Coolant Accident.
The higher Pressurizer level.

B. EOP 2532, Loss of Coolant Accident.
The lower Pressurizer level.

C. EOP 2528, Loss of Forced Flow/Loss of Offsite Power.
The higher Pressurizer level.

D. EOP 2528, Loss of Forced Flow/Loss of Offsite Power.
The lower Pressurizer level.

<<

>>QUESTION 76

K&A Rating: APE 008, Pressurizer Vapor Space Accident AA2.27 (2.9)

K&A Statement: Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: Effects on indicated PZR pressure and/or level sensing line leakage.

Key Answer: **A**

Justification:

A. **CORRECT**: EOP 2532, Loss of Coolant Accident and the higher Pressurizer level is correct. The PORV open is a LOCA and RCPs are not operating due to the loss of Offsite power. Entry into EOP 2532 is correct due to Note 4 on the Diagnostic Flow chart. This note states "The LOCA, SGTR, ESDE, and LOAF events do NOT require offsite power for mitigation. Therefore, if one of these events occurs concurrently with a LOOP, this event should be considered a "single event" and this event's ORP should be implemented in lieu of the LOOP ORP". If the Pressurizer level instrument variable leg had a leak the instrument would indicate lower that actual level. Actual Pressurizer level will rise and fill the Pressurizer because RCS inventory is being discharged out of the PORV.

B. **Incorrect:** Lower Pressurizer level is not correct.
Plausible: A LOCA is a loss of RCS inventory. All LOCAs, except those at the top of the Pressurizer, have low Pressurizer level.

C. **Incorrect**: EOP 2528, Loss of Forced Flow/Loss of Offsite Power is not correct. The Diagnostic Flow chart specifies that if a LOCA and LOOP occur concurrently then the LOCA should be considered a "single event" and the LOCA ORP should be implemented in lieu of the LOOP ORP.
Plausible: A loss of Forced Flow/Loss of Offsite Power has occurred and the higher Pressurizer level is correct.

D. **Incorrect:** Both EOP 2528, Loss of Forced Flow/Loss of Offsite Power and lower Pressurizer level are not correct.
Plausible: A loss of Forced Flow/Loss of Offsite Power has occurred. A LOCA is a loss of RCS inventory. All LOCAs, except those at the top of the Pressurizer, have low Pressurizer level.

<u>SRO Justification</u>: The candidate needs to assess conditions and select an appropriate procedure during an emergency situation. The question 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 mitigating strategy of a procedure.

References: EOP 2541, Appendix 1 Diagnostic Flowchart rev. 002-00, PLC Lesson Text rev. 6

Student Ref: None

Learning Objective: 283657 (05432-MB) Given a set of plant conditions and the Diagnostic Flowchart, identify the most appropriate EOP to mitigate the event.
281907 (02982-MB) Given the plant with a steam bubble in the pressurizer, and given a pressurizer level or pressure transmitter failure (high or low) on either control channel (selected or non-selected), describe: c) The plant response if no operator action is taken.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 76): 08/15/2020 jwr. The NRC commented on 7/24/2020 that the question was possibly not SRO level since it involved a safety function. They suggested putting in SPTA and giving plant conditions and then have them select a procedure. A new K/A was requested and received. A new question was written to address the NRC's comments.

DF pervious to above. Added Rx Vessel Level to stem
<<

>>QUESTION 77

The plant is operating at 100% power.

RCP A ANTIREV BRG TEMP HI (AA-19) alarms.

PPC point T-190*, RCP A ANTI-REV DEV TEMP, is at the alarm setpoint of 195 ºF and stable.

What action is procedurally required?

A. START the "A" RCP Lift Pumps, P-51A/53A i.a.w. ARP 2590B-081, RCP A ANTIREV BRG TEMP HI (C-02/3, AA-19).

B. Declare the COOLANT LOOP inoperable i.a.w. Technical Specification 3.4.1.1, COOLANT LOOPS AND COOLANT CIRCULATION.

C. START a controlled plant shutdown i.a.w. OP 2204, Load Changes.

D. TRIP the Reactor, then TRIP the 'A' and 'C' RCPs. i.a.w. ARP 2590B-081, RCP A ANTIREV BRG TEMP HI (C-02/3, AA-19).

<<

>>QUESTION 77

K&A Rating: APE 015, Reactor Coolant Pump Malfunctions AA2.02 (SRO 3.0)

K&A Statement: Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions: Abnormalities in RCP air vent flow paths and/or oil cooling systems

Key Answer: **A**

Justification:

A. **CORRECT:** The lift pumps are started when this alarm annunciates at 194 °F

B. I**ncorrect:** The RCP has not met a condition in which it is required to be tripped. The LOOP is considered to be OPERABLE with the RCP in operation (Automatic reactor trips will occur if the RCP flow requirements are not met. The ARP does not direct the operator to declare the RCP inoperable.
Plausible: might think loop is inoperable if the RCP is alarming.

C. **Incorrect:** A downpower is commenced if ARD temperature rises to >221 °F
Plausible: action is part of the ARP

D. **Incorrect:** The RCP is tripped if the ARD temperature rises to > 250 °F after tripping the reactor. 'C' RCP is not tripped

Plausible: 'A' RCP is tripped after tripping the reactor. May think trip 2/leave 2 strategy is in effect

SRO Justification: the candidate needs to assess condition s and select an appropriate procedure during an abnormal situation. The question 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 mitigating strategy of a procedure.

References: ARP 2590B-081, RCP A ANTIREV BRG TEMP HI (AA-19)

Student Ref: NONE
Learning Objective: 281983, Describe the purpose of each of the following major RCS components: A) RCP Anti-reverse Device B) RCP Flywheel
Question Source: New
Question History: New
Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 77):
02SEP20 DF – Changed D to read 'TRIP the Reactor, then TRIP…'
Added procedures per NRC comments DF 8/10/20
<<

>>QUESTION 78

During an Excess Steam Demand Event,

1. What Time Credited Operator Action (TCOA) is required to be performed? AND

2. What is the basis for the TCOA?

A. 1. CET temperature is stabilized within five (5) minutes of the affected S/G boiling dry.
   2. Prevents Pressurized Thermal Shock (PTS) of the RCS.

B. 1. Auxiliary Feedwater is isolated to the faulted S/G within thirty (30) minutes of the event.
   2. Prevents Pressurized Thermal Shock (PTS) of the RCS.

C. 1. CET temperature is stabilized within five (5) minutes of the affected S/G boiling dry.
   2. Limits peak Containment Pressure post-accident.

D. 1. Auxiliary Feedwater is isolated to the faulted S/G within thirty (30) minutes of the event.
   2. Limits peak Containment Pressure post-accident. <<

>>QUESTION 78

K&A Rating: Generic, Emergency Procedures /Plan G 2.4.18 (4.0)

K&A Statement: Knowledge of the specific bases for EOPs → 040 Steam Line Rupture

Key Answer: **D**

Justification:

A. **Incorrect:** Stabilizing CET temperatures is not a TCOA. And there is no time requirement associated with stabilizing CET temperatures.
Plausible: Stabilizing CET temperatures is a critical task for simulator evaluations is performed in conjunction with HPSI throttle/stop to prevent PTS.

B. **Incorrect:** AFW is isolated to faulted S/G within 30 minutes of MSIS but not to prevent PTS.
Plausible:Isolating AFW does prevent further Cooldown of the RCS. But it does not prevent PTS.

C. **Incorrect**: Stabilizing CET temperatures is not a TCOA. And there is no time requirement associated with stabilizing CET temperatures. Also it is not performed to limit peak CTMT pressure during a MSLB.
Plausible: Stabilizing CET temperatures is a critical task for simulator evaluations.

D. **CORRECT:** The TCOA action for a MSLB is to Isolate AFW to the affected S/G within 30 minutes of event. This action limits peak CTMT pressure during a MSLB.

SRO Justification: The candidate needs to assess conditions and select an appropriate procedure during an emergency situation. The question 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 mitigating strategy of a procedure.

References: C OP 200.18, Time Critical Action Validation And Verification

Student Ref: None

Learning Objective: 283855, Outline and explain the bases for major actions in EOP 2536, Excess Steam Demand

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 78): 13AUG20 DF. Made question into a 2 part based on comments from NRC<<

>>QUESTION 79

The plant trips from 100% power due to a loss of DV-20.

The crew enters EOP 2525 with the following Steam Generator (SG) conditions:

- #1 SG level is 30% and lowering.
- #2 SG level is 36% and rising.

1. What EOP will you transition to? AND

2. What actions are taken?

A. 1. EOP 2528, Loss of Offsite Power/Loss of Forced Circulation.
   2. Place the 'B' Auxiliary Feedwater Regulating Valve in local manual and throttle flow.

B. 1. EOP 2528, Loss of Offsite Power/Loss of Forced Circulation.
   2. Throttle the Feed Regulating Valve Bypass Valves.

C. 1. EOP 2526, Reactor Trip Recovery.
   2. Place the 'B' Auxiliary Feedwater Regulating Valve in local manual and throttle flow.

D. 1. EOP 2526, Reactor Trip Recovery.
   2. Throttle the Feed Regulating Valve Bypass Valves.

<<

>>QUESTION 79

K&A Rating: APE 054, Loss of Main Feedwater AA 2.06 (4.3)

K&A Statement: Ability to determine and interpret the following as they apply to the Loss of Main Feedwater: AFW adjustments needed to maintain proper T-ave and S/G level

Key Answer: **C**

Justification:

A. **Incorrect:** EOP 2528 is not correct. Two RCPs will still be able to be operated (will still have RBCCW flow) and therefore the RCS will remain in forced circulation.
Plausible: The examinee may think that RCP power is lost with a loss of DV-20 and the trip of the plant because the busses that feed the RCPs do not transfer to the RSST. DV power is used to operate breakers and on the trip breakers must operate to transfer power from one bus to another.

B. **Incorrect:** EOP 2528 is not correct. Two RCPs will still be able to be operated (will still have RBCCW flow) and therefore the RCS will remain in forced circulation. And throttling the Feed Regulating Valve Bypass Valves is not correct because the loss of DV-20 will cause a trip of the Main Feedwater pump on low vacuum due to the loss of the condenser.
Plausible: The examinee may think that RCP power is lost with a loss of DV-20 and the trip of the plant because the busses that feed the RCPs do not transfer to the RSST and may not remember that a loss of DV-20 results in the loss of the condenser.

C. **CORRECT:** Entry to EOP 2526 and place the 'B' Auxiliary Feedwater Regulating Valve in local manual and throttle flow are correct. The Loss of DV-20 closes the MSIVs and results in the loss of Main Feedwater pumps. AFW will be used to feed the SGs in EOP 2525 and if the feed rate is not sufficient, AFW will be throttled to maintain SG levels. Two RCPs will be in operation therefore EOP 2526 is the appropriate EOP to transition to.

D. **Incorrect**: Throttling the Feed Regulating Valve Bypass Valves is not correct because the loss of DV-20 will cause a trip of the Main Feedwater pump on low vacuum due to the loss of the condenser.
Plausible: Examinee may not remember that a loss of DV-20 results in the loss of the condenser.

<u>SRO Justification</u>: The candidate needs to assess conditions and select an appropriate procedure during an abnormal situation. The question 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 mitigating strategy of a procedure.

References: AOP 2506B, Loss of Vital 125 VDC Instrument Panel DV20

Student Ref: None

Learning Objective: 283684 - Given a set of plant conditions, determine the action required in accordance with the Instructions and Contingency Actions of EOP 2526, Reactor Trip Recovery

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 79): 13AUG20 DF added requirement to determine what procedure to transition to based on NRC comments.

<<

>>QUESTION 80

The plant is in MODE 5 cooling down for a refuel outage with T_{AVE} = 185 °F.

2-SI-306 and 2-SI-657 are in remote control.

A transient occurs with the following indications:

- INVERTER INV-1 TROUBLE (C-08, A-25).
- VA-10 ON BYPASS SOURCE INV-5 (C-08, A-26).
- INVERTER INV-5 TROUBLE (C-08, A-27).
- RPS Channel 'A' is dark.

1. What procedure addresses the current plant situation? AND

2. What actions does the procedure direct to mitigate the consequences of this event?

A. 1. AOP 2504C, Loss of 120 VAC Vital Instrument Panel VA-10.
   2. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve in manual control and throttle open.

B. 1. AOP 2572, Loss of Shutdown Cooling.
   2. Place 2-SI-657, SDC Heat Exchanger Flow Control Valve in manual control and throttle open.

C. 1. AOP 2504C, Loss of 120 VAC Vital Instrument Panel VA-10.
   2. Place 2-SI-306, SDC Total Flow Control Valve in manual control and throttle open.

D. 1. AOP 2572, Loss of Shutdown Cooling.
   2. Place 2-SI-306, SDC Total Flow Control Valve in manual control and throttle open.
   .

<<

>>QUESTION 80

K&A Rating: APE 057 G2.1.7 (SRO 4.7)

K&A Statement: APE 057 Loss of Vital AC Instrument Bus 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Key Answer: **B**

Justification: The indications given in the stem are those of a loss of VA-10. Not other indications are given for a loss of all power. During a loss of VA-10, SI-306 fails open and SI-657 fails closed. The SDC pumps will be pumping RCS, bypassing the SDC heat exchangers, heating up the RCS. The actions to mitigate this condition are found in AOP 2572, Loss of SDC, Attachment J & K. SI -306 and 657 are placed in local control and throttled to pre-event positions (306 throttled closed, 657 throttled open)

A. **Incorrect:** AOP 2504C would be entered but directs the operator to AOP 2572 to Restore Shutdown Cooling There are no steps in AOP 2504C to place the SDC valves in manual
Plausible: recognizes VA-10 is lost, may think actions to address loss of SDC can be found there

B. **CORRECT:** AOP 2572 would be appropriate for a loss of VA-10, has the SDC responder be directed to take manual control of SI-657 and open the valve to allow cooling of the RCS

C. **Incorrect:** AOP 2504C would be entered but directs the operator to AOP 2572 to Restore Shutdown Cooling . There are no steps in AOP 2504C to place the SDC valves in manual
Plausible: recognizes VA-10 is lost, may think actions to address loss of SDC can be found there

D. **Incorrect**: AOP 2572 woud be entered but SI-306 wouldn't be throttled open since the loss of VA-10 fails the valve open
Plausible: May think SI-306 is throttled open to re-establish SDC

<u>SRO Justification</u>: The candidate needs to know conditions and limitations contained in the facility license. The question 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 mitigating strategy of a procedure.

References: AOP 2572, Loss of Shutdown Cooling
AOP 2583, Loss of All AC Power During Shutdown Conditions

Student Ref: NONE

Learning Objective: 283351 Given a set of plant conditions, determine the section within AOP 2572, Loss of Shutdown Cooling the best mitigates the abnormal conditoin

Question Source: New
Question History: New
Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 80):

removed part 1 of 2 part question df 6/1/20.

Total re-write 6/9/20. Changed 3.0.5 to 3.0.3 in response to NRC comments DF 8/10/20

Total re-write 14AUG20 due to question not meeting the K/A

20AUG20 DF - swapped Total Loss of AC with Loss of VA-10 based on NRC comments. Reworde stem to be more specific regarding **procedurally directed actions**.

<<

>>QUESTION 81

Both Millstone Units are at 100% power.

The Turbine Driven Auxiliary Feedwater (TDAFW) Pump is out of service for maintenance and the applicable TSAS has been entered.

Then, ISO-New England/CONVEX declares a Capacity Deficiency Alert, and predicts that switchyard voltage would drop to 330 kV if either Millstone unit was lost.

C OP 200.8, "Response to ISO/CONVEX Alerts", has been entered and appropriate actions are being taken.

Which of the following describes the change in Technical Specification Action Statement, based on the given conditions?

A. ONLY TSAS LCO 3.8.1.1 action d. is entered because 2 Offsite Circuits are INOPERABLE.

B. ONLY TSAS LCO 3.8.2.1 is entered because buses 24C AND 24D are INOPERABLE.

C. ONLY TSAS LCO 3.0.3 is entered because 2 Offsite Circuits AND the TDAFW pump are INOPERABLE.

D. BOTH TSAS LCO 3.8.1.1 action d. AND TSAS 3.0.3 are entered because 2 Offsite Circuits AND the TDAFW pump are INOPERABLE.

<<

>>QUESTION 81

K&A Rating: Generic, APE 077, Generator Voltage and Electric Grid Disturbances (SRO 3.9)
K&A Statement: G2.1.28 Knowledge of the purpose and function of major system components and controls.

Key Answer: **A**

Justification:

A. **CORRECT:** If RSST voltage decreases to less than 3,900 volts, per SP 2619G-001, "One Offsite Circuit Inoperable" the RSST is inoperable. Per C OP 200.8, if ISO New England predicts grid voltage will drop below 345 KVA on loss of either Millstone Unit, both offsite lines are considered inoperable. Therefore, TSAS 3.8.1.1, Action d should be entered.

B. **Incorrect:** TS 3.8.2.1 gives action requirements for "less than the above compliment of A.C. Busses" being OPERABLE, which would cover more than one vital VAC bus being out.
Plausible:: Student may remember busses 24C and 24D are required to be OPERABLE and believe that the loss of both vital AC bus operability would clearly not meet the TSAS.

C. **Incorrect**: TS 3.8.1.1 has a requirement for additional action of the loss of the TDAFW pump only if an emergency diesel generator is also inoperable.
Plausible: Student may remember the requirement for the TDAFW pump in the offsite power TSAS but not how it relates to the applicability of the EDG requirement.

D. **Incorrect:** The loss of the TDAFW pump is only considered an additional vulnerability by the Offsite Power Tech Spec if one of the EDGs is not available.
Plausible: Student may considering the additional plant vulnerability and believe because the TS 3.8.1.1 does not cover a loss of the TDAFW pump with the loss of two offsite power sources, this must require TS 3.0.3 be applied.

<u>SRO Justification</u>: This question is SRO only as it requires assessing plant conditions and determining how changing conditions affect notification requirements and applicable TS action requirements of > 1 hour. The question 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 mitigating strategy of a procedure. This question is linked to 10 CFR 55.43(b)(2).

References: TS LCO 3.8.1.1 Student Ref: NONE
Learning Objective:
Question Source: Bank
Question History: Millstone 2018 NRC exam – question 81
Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X
10CFR55: CFR55.43 (b)(2)
Comments (Question 81): 6/29/20 jwr. Replaced K/A and question. Minor editing to read easier DF 8/10/20

<<

>>QUESTION 82

The plant is operating at 100% power. A transient occurs with the following indications:

- Buses 25A, 24A, and 24C breaker indicating lights are not lit (C-08).

A plant trip occurs and the following is observed:

- RSS SPLY BKR 22S2-25B-2 (H204), GREEN indicating light is lit (C-08).

What EOP will be entered following the completion of EOP 2525, Standard Post Trip Actions?

A. EOP 2526, Reactor Trip Recovery.

B. EOP 2528, Loss of Off-Site Power/Loss of Forced Circulation.

C. EOP 2530, Station Blackout.

D. EOP 2537, Loss of All Feedwater.

<<

>>QUESTION 82

K&A Rating: APE A13, Natural Circulation Operations (SRO 3.7)

K&A Statement: AA2 Ability to determine and interpret the following as they apply to the (Natural Circulations Operations): AA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Key Answer: **B**

Justification: Due to limited EOP options in CE procedure set, this question was determined the best way to meet/address the K/A.

A. **Incorrect:** No RCPs are operating, Core Heat removal is compromised
Plausible: might think LOOP/LOFC did not occur since have Facility 2 4160 busses

B. **CORRECT:** Losing bus 201A causes bus D11 to de-energize. D11 powers the Channel 1 Auto-xfer from NSST to RSST. With bus 25B failing to transfer, no RCPs will be operating post-trip. With all other safety functions being met, the diagnostic will direct the operator to EOP 2528**.**

C. **Incorrect**: Not in a Station Blackout, Facility 2 4160 busses are energized.
Plausible: Might think all busses lost power

D. **Incorrect:** All Feedwater is not lost. Still have TDAFW pump operated from DV20 and "B" AFW pump via Bus 24D.
Plausible: Losing Bus 201A takes away both Main Feedwater pumps, "A" and "B" Condensate pumps, and "A" AFW pump. Loss of 25B takes away the "C" Condensate pump. Examinee could also conclude "B" AFW pump is lost as well as TDAFW pump control power.

<u>SRO Justification</u>: The candidate needs to assess conditions and select an appropriate procedure during an abnormal situation. The question cannot be answered by solely knowing system knowledge, immediate operator actions, AOP or EOP entry conditions, or the purpose, overall sequence of events, or overall mitigating strategy of a procedure.

References: EOP 2541, Appendix 1, Diagnostic Flowchart
AOP 2505A, Loss of Vital 125 VDC Bus 201A

Student Ref: NONE

Learning Objective: 282877 Evaluate abnormal Vital 125VDC distribution system operation and determine the cause and course of action iaw AOP 2505A

Question Source: 56600

Question History: 2004 Audit Exam

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: CFR 55.43 (b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments (Question 82): 6/9/2020 Changed RWST volume to get away from confusing temperature requirements. 7/1/20 DF. Got new K/A from D. Silk because a question could not be written to meet the original K/A..

14AUG20 DF Added to justification words to explain why thquestion was reasonable for SRO level question → CE procedure set has limited EOP options.

<<

>> QUESTION 83

The following alarms have annunciated due to a radioactive gas leak:

- PROCESS MON RAD HI HI/FAIL (C-06/7, DA-24).

1. The leak can be located by having Health Physics sample the _____. AND
2. ARP _____ is used to provide actions to address the event.

A. 1. -25' Auxiliary Building general area.
   2. 2590E-135, WASTE GAS DISCHARGE TO UNIT 1 STACK RIT-9095.

B. 1. 14' East Penetration Room.
   2. 2590E-135, WASTE GAS DISCHARGE TO UNIT 1 STACK RIT-9095.

C. 1. -25' Auxiliary Building general area.
   2. 2590H-031, UNIT 2 STACK GASEOUS RIT-8132B.

D. 1. 14' East Penetration Room.
   2. 2590H-031, UNIT 2 STACK GASEOUS RIT-8132B.

<<

>>QUESTION 83

K&A Rating: APE 060, Accidental Gaseous Radwaste Release (SRO 4.0)

K&A Statement: AA2.02 The possible location of a radioactive-gas leak, with the assistance of a PEO, health physics and chemistry personnel

Key Answer: **C**

Justification: The PROCESS MON RAD HI HI/FAIL (C-06/7, DA-24) annunciator is a common alarm, when it alarms, the crew is directed to determine what radiation monitor is alarming. If a radioactive gas leak occurs, there are limited sources where it can emanate from. The -25' general area contains the Waste Gas Tank farm, a likely source for a gas leak. Any gas leak would be to the Aux building atmosphere and directed to the Unit 2 stack through the Aux Bldg ventilation system and detected by the Unit 2 Stack Rad Monitor, RM-8132

A. **Incorrect:** RM-9095 is only in service during a Radioactive gas discharge.
Plausible: Might think Waste Gas is source of leak and 9095 is a good RM to use.

B. **Incorrect:** RM-9095 is only in service during a Radioactive gas discharge.
Plausible: The 14'6 East penetration Room is above the -5 West piping penetration room has letdown, charging , and other piping and any gaseous leak would rise into other areas. Examinee needs to understand what area have what equipment.

C. **CORRECT:** Good location and proper procedure

D. **Incorrect**: 14' East penetration Room has no gas source for leak
Plausible: The 14'6 East penetration Room is above the -5 West piping penetration room has letdown, charging , and other piping and any gaseous leak would rise into other areas. Examinee needs to understand what area have what equipment.

<u>SRO Justification</u>: The candidate needs to understand radiation hazards that may arise during normal situations. The question 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 mitigating strategy of a procedure.

References: 25203-26029 P&ID Auxiliary Building Ventilation System sh. 2
ARP 2590H-055

Student Ref: NONE

Learning Objective:

Question Source: New
Question History: New
Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: 10CFR55(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions

Comments (Question 83): 09/10/2020 jwr. Don Jackson's feedback on 09/10/2020 was that this was not and SRO level question. Their suggestion was to make a 2X2 with an area and what procedure would be used. A question was written to address their feedback.

Changed 'C' to -25 general area. <<

>>QUESTION 84

The plant has been shut down for refueling.

- RCS temperature = 221 ºF lowering.

What condition is permitted by Technical Specifications?

A. The Equipment Hatch is closed with 4 bolts.

B. Containment Purge is in operation.

C. The Containment air lock interlock is defeated. The inner door is closed.

D. 2-SI-463, SITs Recirc Header Stop, is open with a Dedicated Operator stationed.

<<

>>QUESTION 84

K&A Rating: APE 069, Loss of Containment Integrity (SRO 4.3)

K&A Statement: AA2.01 Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Loss of Containment Integrity

Key Answer: **D**

Justification:

A. **Incorrect:** Equipment Hatch is required to be closed and sealed. 4 bolts is good for CTMT Closure, not CTMT INTEGRITY
Plausible: might confuse CTMT Closure with CTMT Integrity

B. **Incorrect:** CTMT Purge is not allowed in Modes 1-4
Plausible: Might confuse Purging CTMT with Venting CTMT

C. **Incorrect**: The interlock cannot be defeated in MODES 1-4. One air lock door needs to be locked closed if interlock is not OPERABLE unless the inoperable door is being worked on.
Plausible: might think door being closed is acceptable

D. **CORRECT:** 2-SI-463 is a manual valve that needs a Dedicated operator to perform TS function to close penetration (still in MODE 4)

SRO Justification: the candidate needs to know conditions and limitations contained in the facility license. The question 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 mitigating strategy of a procedure.

References: Tech Spec 3.6.1.1

Student Ref: NONE

Learning Objective: 280982, As defined in the Unit 2 Technical Specifications, state the conditions which constitute:

A. Containment Integrity

B. Enclosure Building Operability

Question Source: New

Question History: New (see South Texas Project 2009 exam question 89)

Cognitive Level: Memory/Fundamental Knowledge:

Comprehensive/Analysis: X

10CFR55: 10CFR43(b)(1) Conditions and limitations in the facility license

Comments: 09/02/2020 dmf. Don Jackson identified that “condition” was mis-spelled in the stem. Fixed mis-spelling.

SRO knowledge of technical specifications to determine if Surveillance requirements for Containment Integrity are being met. Changed stem in response to NRC comments, added information to justification DF 8/10/2020

<<

>>QUESTION 85

The plant is in MODE 3.

A reactor startup is in progress per OP 2202, Reactor Startup ICCE.

- T_{AVE}/T_{REF} (C-04, DA-19) annunciates.
- T_{COLD} is 495 ºF and lowering.

What are the procedurally directed actions?

A. Secure diluting the RCS; take actions to raise T_{AVE} to greater than 515 ºF per OP 2202, Reactor Startup ICCE

B. Trip the reactor and initiate emergency boration per AOP 2558, Emergency Boration.

C. Trip the reactor and stop two Reactor Coolant Pumps (RCPs) per EOP 2525, Standard Post Trip Actions.

D. GO TO OP 2302A, Control Element Drive System and insert all CEAs in sequence.

<<

>>QUESTION 85

K&A Rating: APE 011, RCS Overcooling (SRO 3.3)

K&A Statement: AA2.1 Ability to determine and interpret the following as they apply to the (RCS Overcooling): Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Key Answer: **B**

Justification: If an uncontrolled cooldown (T_{COLD} less than 500 ºF) occurs, the operator is directed to trip the reactor, secure a Reactor Coolant Pump, and initiatiate emergency boration iaw AOP 2558, Emergency Boration (OP 2202, Reactor Startup ICCE, Attachent 5, step 3)

A. **Incorrect:** Trip criteria met iaw OP 2202
Plausible: These actions are taken when T_{AVE} < 515 ºF

B. **CORRECT:** Correct actions per OP 2202 conditional actions.

C. **Incorrect:** Only one RCP is secured (< 500 ºF).
Plausible: might think "trip two/leave two" strategy would be used

D. **Incorrect:** An uncontrolled cooldown indicates a loss of control. The prudent (and procedurally required) action would be to trip the reactor..
Plausible: OP 2202 has actions to insert CEAs upon termination of start up.

SRO Justification: the candidate needs to know conditions and limitations contained in the facility license. The question 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 mitigating strategy of a procedure.

References: OP 2202, Reactor Start Up ICCE, Attachment 5, Conditional Actions

Student Ref: NONE

Learning Objective: 282473, Based on the given plant conditions and an operational situation involving a reactor startup, describe any required actions per OP 2202, Attachment 5, Reactor Start up Conditional Actions

Question Source: question 1154362
Question History: Modified
Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 10CFR43(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments SRO assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations. Added procedures to A and C in response to NRC comments DF 8/10/2020

<<

>>QUESTION 86

A reactor startup is in progress with the following conditions:

- RCS temperature is 532 °F.
- Pressurizer pressure is 2250 psia.
- The Reactor is critical with power stable at 1E-3%.

Temperature (°F)	RCP P40A	RCP P40B	RCP P40C	RCP P40D
Stator Winding	171	178	225	172
Upper Guide Bearing	149	197	150	150
Lower Guide Bearing	163	173	154	151
Upper Thrust Bearing	132	149	127	157
Lower thrust Bearing	130	135	115	125
Anti-Reverse Bearing	152	189	162	156

1. What procedure provides direction for this condition? AND

2. What actions are directed?

A. 1. ARP 2590B-130, RCP C STR TEMP HI.
   2. Trip the Reactor, and stop the “C” RCP.

B. 1. ARP 2590B-124, RCP B UPPER GUIDE TEMP HI.
   2. Trip the Reactor, and stop the “B” RCP.

C. 1. OP 2202 Reactor Startup ICCE.
   2. Terminate the reactor startup by inserting Reg. Group CEAs, and stop the “C” RCP.

D. 1. OP 2202 Reactor Startup ICCE.
   2. Terminate the reactor startup by inserting Reg. Group CEAs, and stop the “B” RCP.

<<

>>QUESTION 86

K&A Rating: 003A2.03 (3.1)

K&A Statement: 003A2.03 Reactor Coolant Pump System. Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems

Key Answer: **B**

Justification:

A. **Incorrect:** The trip of the "C" RCP is not required.
Plausible: Tripping the Reactor and Turbine and stopping the "B" RCP is correct. Examinee may thing two RCPs must be secured since in accident conditions we trip two RCPs, i.e trip two leave two for SIAS actuation.

B. **CORRECT**: The ARP specifies if "RCP B UPR GUIDE BRG TEMP" is greater than 194 °F and is validated to be above 194 °F, then TRIP Reactor, TRIP Turbine, STOP "B" RCP, and Go To EOP 2525, "Standard Post Trip Actions".

C. **Incorrect**: The ARP specifies if greater than 194 °F and is validated to be above 194 °F, then TRIP Reactor, TRIP Turbine, STOP "B" RCP. The ARP actions would be the priority guidance for the US.
Plausible: Examinee may believe OP 2202, Reactor Startup, is the priority procedure since it is an ICCE and OP 2202 does have guidance to terminate for actual or potential equipment damage, and terminates by inserting Reg. Group CEAs. Examinee may also think two RCPs must be secured since in accident conditions we trip two RCPs, i.e. trip two leave two for SIAS actuation.

D. **Incorrect:** The ARP specifies if greater than 194 °F and is validated to be above 194 °F, then TRIP Reactor, TRIP Turbine, STOP "B" RCP. The ARP actions would be the priority guidance for the US.
Plausible: Examinee may believe OP 2202, Reactor Startup, is the priority procedure since it is an ICCE and OP 2202 does have guidance to terminate for actual or potential equipment damage, and terminates by inserting Reg. Group CEAs. And stopping the "B" RCP is correct.

<u>SRO Only Justification</u>: This question is SRO only as it requires assessing plant conditions and then selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess a plant condition (high temperature on a RCP bearing) and select appropriate procedures (ARP or OP actions which requires a plant trip and stopping the RCP).

References: ARP 2590B-124 Rev 002, OP 2202 Rev 025 Student Ref: NONE

Learning Objective: 286064 Given a set of plant conditions determine if any RCP trip criteria is met.

Question Source: New written from Bank (451719, 5000049-MB)

Question History: Q-92 on the NRC 2005 exam, 003/A2.03. Question answers have all been changed. Changed stem, added detailed to justifications, added SRO justification, added references, and added learning objective.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 86):
02SEP20 DF – Added table of RCP parameters in response to Don Jackson's review.

8/10/20 jwr. Completed changed recommended by the NRC 07/24/20. The NRC recommended that answers "B" and "D" be changed to add "only" "B" RCP to make it clear that only "B" RCP is stopped. The word "ONLY" was answers "B" and "D" and the word "BOTH" was added to answers "A" and "C".

06/23/20 jwr. Change answers "A" and "B" based on Pat S comment that tripping the turbine is not required, and that is why he missed question. 06/22/20 jwr. Changed question because it was a high hit. Wrote new 2X2 question from bank question. Less system knowledge and more SRO procedure knowledge.
<<

>>QUESTION 87

The plant is operating at 100% power with bus 24E aligned to bus 24D.

Due to problems with the “A” RBCCW Pump the Unit Supervisor directed entry to AOP 2564, Loss of RBCCW procedure.

- The “B” RBCCW pump was started.
- The “A” RBCCW pump was stopped (handswitch not in Pull to Lock) and declared Inoperable.

The following alarms are in on Main Control Board CO-6/7:

- RBCCW PUMPS MISALIGNED.
- RBCCW PUMP B SIAS/LNP START MANUALLY BLOCKED.

1. How will OPERABILITY of the “A” RBCCW header be restored, while maintaining the plant at 100% power? AND
2. What pumps would automatically start on a SIAS/LNP?

A. 1. The “A” RBCCW pump must be repaired.
   2. The “B” and “C” RBCCW pumps would restart on a SIAS/LNP.

B. 1. Swap 24E to 24C and place the “SIAS/LNP Actuation Signal Handswitch 6119D to “Normal”.
   2. The “B” and “C” RBCCW pumps would restart on a SIAS/LNP.

C. 1. The “A” RBCCW pump must be repaired.
   2. The “A” and “C” RBCCW pumps would restart on a SIAS/LNP.

D. 1. Swap 24E to 24C and place the “SIAS/LNP Actuation Signal Handswitch 6119D to “Normal”.
   2. The “A” and “C” RBCCW pumps would restart on a SIAS/LNP.

<<

>>QUESTION 87

K&A Rating: 008G2.4.46 (4.2)

K&A Statement: 008G2.4.46 Component Cooling Water System, Emergency Procedure/Plan: Ability to verify that the alarms are consistent with the plant conditions.

Key Answer: **C**

Justification:

A. **Incorrect:** The "A" and "C" RBCCW pumps will restart on a SIAS/LNP not the "B" and "C". The "B" RBCCW pump is blocked from starting so that the "B" Emergency Diesel Generator is not overloaded.
Plausible: It is reasonable to believe that the running pump would restart and the pump that was off would not restart. The student must understand what the alarms that are in mean.

B. **Incorrect:** Swapping busses is not correct. Swapping busses would require de-energizing 24E which would stop the "B" RBCCW pump for greater than 5 minutes and that would require a plant trip.
Plausible: Swapping 24E to 24C and placing the "SIAS/LNP Actuation Signal Handswitch 6119D to "Normal" does restore the "A" RBCCW header to Operable. However it doesn't meet the stem condition to maintain the plant at 100% power. The "A" and "C" RBCCW pumps will restart is correct.

C. **CORRECT:** The "A" RBCCW pump must be repaired to restore the ""A RBCCW header to Operable because swapping bus 24E to 24C would require a plant shutdown. The "A" and "C" RBCCW pumps will restart on a SIAS/LNP is correct. The RBCCW PUMPS MISALIGNED alarm is in and this indicates that the "A" RBCCW pump is not in PTL and will therefore start on a SIAS/LNP signal. The RBCCW PUMP B SIAS/LNP START MANUALLY BLOCKED is in so the "B" RBCCW pump will not start on a SIAS/LNP.

D. **Incorrect**: Swapping busses is not correct. Swapping busses would require de-energizing 24E which would stop the "B" RBCCW pump for greater than 5 minutes and that would require a plant trip.
Plausible: The "A" and "C" RBCCW pumps will restart is correct. Swapping 24E to 24C and placing the "SIAS/LNP Actuation Signal Handswitch 6119D to "Normal" does restore the "A" RBCCW header to Operable. However it doesn't meet the stem condition to maintain the plant at 100% power.

<u>SRO Only Justification</u>: This question is SRO only as it requires assessing plant conditions and applying knowledge of Technical Specification bases to analyze TS required actions. The question cannot be answered by solely knowing systems knowledge, or LCO information listed in the LCO statement. The candidate must understand the condition of the RBCCW system based on the alarms in and understand the bases for the Operability of the RBCCW system relative to separate power supplies.

References: ARP 2590E-097 Rev 000-01, ARP 2590E-064, OP 2330A Rev 028-00, AOP 2564 Rev 005-00, AOP 2585 Rev 003-00, TS Bases 3/4.7.3 Rev change No. 397
Student Ref: NONE

Learning Objective: 281951 (03020-MB) Given any operating condition for the RBCCW System, state whether the condition requires entry into the Technical Specification.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 43(b)(2) Facility operating limitations in the technical specifications and their bases

Comments (Question 87): 08/10/20 jwr. Completed NRC changes recommended on 7/24/20. The NRC recommended that a space be added in the stem in the second bullet after “stopped” and the order of the question be switched to put how will OPERABILITY… first and then what pumps would automatically start second. These changes were made as well as reorder answer justifications in the stem.

6/1/20 jwr. PF commented that the “A” service pump returned to service was confusing since it would auto start. Removed returned to service to address comment.
<<

>>QUESTION 88

The plant is at 100% power and Forcing Pressurizer Sprays.

Then, VR-21 is lost causing all Backup Heaters to trip.

Reactor Coolant system pressure (RCS) lowers to 2200 psia. After manipulations of the operational pressure control components the RCS pressure begins to rise.

VR-21 remains deenergized.

What is the required Technical Specification action for the above plant conditions?

A. Restore RCS pressure to within its limits within two (2) hours.

B. Restore at least two (2) groups of Backup Heaters within 72 hours.

C. Restore CEAPDS reed position indication within (4) hours.

D. Restore the inoperable bus to operable status within eight (8) hours.

<<

>>QUESTION 88

K&A Rating: 010G2.2.22 (4.7)

K&A Statement: 010G2.2.22 (4.7) Pressurizer Pressure Control system: General Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Key Answer: **A**

Justification:

A. **CORRECT:** One of the ways the DNB margin is preserved is by maintaining RCS pressure greater than 2225 psia. If Pressurizer pressure lowers to < 2225 psia, TS 3.2.6 requires that pressure be restored to > 2225 psia within 2 hours or THERMAL POWER be reduce to < 5% of THERMAL POWER within the next 4 hours.

B. **Incorrect:** TS 3.4.4 requires at least two groups of heaters. The TS Bases specifies that the heaters required are Proportional Heaters since they must be powered from an emergency 480V electrical bus.
Plausible: TS 3.4.4 states that "at least two groups of pressurizer heaters each having a capacity of at least 130 KW" be OPERABLE. It is reasonable the candidate could think this is correct if they don't know the Bases. The requirement to restore the pressurizer heaters within 72 hours is correct.

C. **Incorrect:** CEAPDS, including Technical Specification required reed switches, remains powered and Operable with the loss of VR-21 because it is feed by both VR-11 and VR-21.
Plausible: CEAPDS is powered from VR-21. CEAPDS reed switch position indication is required to meet TS 3.1.3.3. The lost of CEAPDS requires entry to TS 3.1.3.3, with an inoperable position indicator channel partially inserted, and restoration within (4) hours.

D. **Incorrect**: VR-21 is one of the available power supplies to power vital bus VA-40. Although the "backup" power supply to a TS required vital power supply has been lost, this does not, in and of itself, make the applicable vital power supply inoperable.
Plausible: VR-21 can supply vital bus VA-40. And VR-21 can be supplied by emergency 480V buses B62 and B61. In addition VR-21 powers many instruments on safety systems. Therefore it is reasonable this in a TS electrical bus. Also the applicable action, restore the inoperable bus to operable status within eight (8) hours, is correct for a vital 120V bus.

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and applying knowledge of Technical Specification bases to analyze TS required actions. The question cannot be answered by solely knowing systems knowledge, or LCO information listed in the LCO statement.

References: TS 3.2.6, TS 3.4.4, TS 3.4.4 Bases, TS 3.8.2.1, TS 3.1.3.3 TS Rev through change No. 397, AOP 2504B Rev 008

Student Ref: NONE

Learning Objective: 281999 (03061-MB) Given a list of plant conditions and/or parameter values, and a copy of Technical Specifications, determine if any Reactor Coolant System LCOs or safety limits are violated and identify appropriate action statement.

Question Source: Bank

Question History: Bank 451626 (3000001-MB) 2003 USRO NRC exam Q-3. Rewrote justification to strengthen correct, incorrect, and plausibly. Replaced answer “C” with a greater than (1) hour TS.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 43(b)(2) Facility operating limitations in the technical specifications and their bases

Comments (Question 88): 09/10/2020 jwr. Changed pressure in stem to 2200 psia based on Angelo Leone feedback.
08/10/20 jwr. Completed change recommended by the NRC on 07/24/20. The NRC recommended that answer “C” be replaced with a greater than (1) TS. Replaced answer “C” with a (4) hour CEAPDS answer.
<<

>>QUESTION 89

The plant has experienced a Loss of Coolant Accident and the following conditions exist:

- Sump Recirculation has actuated.
- The Safety Injection Recirculation Header Isolation valves, 2-SI-659 and 2-SI-660 have NOT automatically positioned.

Which one of the following describes WHEN the Unit Supervisor would direct these valves to be closed?

A. Only after RWST header isolation valves (2-CS-13.1A & 2-CS-13.1B) are closed.

B. Immediately after verifying 30 gpm minimum flow from each HPSI pump.

C. Immediately after other SRAS actuations have been verified.

D. Only after overriding and securing both LPSI pumps.

<<

>>QUESTION 89

K&A Rating: 013A2.03 (4.7)

K&A Statement: 013A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rapid depressurization.

Key Answer: **C**

Justification:

A. **Incorrect:** The RWST header isolation valves are closed in a subsequent step. Closing the RWST header isolation valves is not part of the SRAS initiation criteria and therefore is not correct.
Plausible: The RWST header isolation valves are closed in a subsequent step and it is reasonable that they could be closed in the SRAS initiation criteria step since once the containment sump outlet valves are closed this isolates the RWST as a suction for the SI and CS pumps. And it is reasonable that you would not want to lose containment sump inventory back to the RWST where you can't get it back.

B. **Incorrect**: HPSI pump minimum flow is verified to be greater 30 gpm in a subsequent step. The HPSI pump minimum flow check is not part of the SRAS initiation criteria and therefore is not correct.
Plausible: It is reasonable that you would verify minimum flow requirements prior to isolating the minimum flow lines and there is a subsequent step that does check minimum flow for the HPSI pumps. And the minimum flow does prevent the pump from overheating.

C. **CORRECT:** Immediately after other SRAS actuations have been verified is correct. The LOCA procedure, under the SRAS Initiation Criteria step, has ensuring the minimum flow valves closed as the last lettered action of this procedure step. Therefore it done last. The procedure Technical Guide specifies the basis of this step is to ensure that the HPSI pump minimum recirculation flow valves are closed to prevent contaminated containment sump water from returning to the RWST.

D. **Incorrect**: Overriding and securing both LPSI pumps is not correct since the LPSI pumps automatically secured as part of the SRAS actuation.
Plausible: Overriding and securing both LPSI pumps would have to be done prior to closing the minimum flow valves since the LPSI pumps are before the minimum flow valve it the SRAS initiation step. And if the LPSI pump were running during a SRAS for any length of time they could affect the HPSI and CS pumps minimum flows due to the hotter suction source.

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selection of appropriate procedures during normal, abnormal, and emergency situations. The student must understand what automatic actions occur in a SRAS and the bases for those actions. They must understand what action to take when the plant does not respond as expected, the sequence of steps and the reasons that the steps are sequenced as they are.

References: EOP 2532 Rev 035-00, EOP 2532 Technical Guide Rev 29
Student Ref: NONE

Learning Objective: 283790 (05941-MB) Explain the basis for the following as specified in EOP 2532 Loss of Primary Coolant: Initiation of SRAS.

Question Source: Bank (451631, 3071905 –MB)

Question History: Question was on the 2003 NRC USRO exam (Q-8, K/A 013A2.01). The question was modified as follows; stem was shorted to remove information about actuation of other SRAS components, answers were reformatted to have second part on second line, answer "A" "why" was changed to be more plausible, and answer justifications were completely rewritten to be clearer.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 89): 08/10/20 jwr. Completed changes recommended by the NRC on 7/24/20. The NRC recommended that the second part of the question be removed from the stem and answers. The question was changed to remove "and WHY" from the stem and the second part of all answers. There was discussion of the K/A match and after discussion the NRC agreed the question was a K/A match.
<<

>>QUESTION 90

Plant startup is in progress under OP 2201, Plant Heatup.

- RCS average temperature is 220 °F.
- “A” and “C” Containment Air Recirculation (CAR) fans are operating in “Fast” speed.
- “B” and “D” CAR fans are not operating.
- Containment temperature is 80 °F and slowly rising.

Component Engineering reports that vibration readings on the “A” CAR fan are significantly higher than normal.

The “A” CAR fan is declared inoperable and the fan is secured per 2314A.

Which one of the following is the impact on the plant startup <u>and</u> what actions are required?

A. Heatup can continue to normal operating temperature.
No actions required. CAR Fans are not required in the current MODE.

B. Heatup can continue to normal operating temperature.
Restore “A” CAR to OPERABLE within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

C. Heatup can continue – temperature can be raised to just below 300 °F.
No actions required. CAR Fans are not required in the current MODE.

D. Heatup can continue – temperature can be raised to just below 300 °F.
Restore “A” CAR to OPERABLE within 48 hours or be in COLD SHUTDOWN within the next 36 hours.

<<

>>QUESTION 90

K&A Rating: 022A2.02 (2.6)

K&A Statement: 022A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS: and (b) based on those predictions, use procedure to correct, control, or mitigate the consequences of those malfunctions or operations: Fan motor vibrations.

Key Answer: **C**

Justification:

A. **Incorrect:** Heatup to normal operating temperature is not correct. The heatup can continue up to just below < 300 °F. The applicable TS 3.6.2.1 is applicable in MODES 1, 2, and 3*. Therefore the TS is not applicable until temperature is greater than or equal to 300 °F. **Plausible:** Examinee may not know the CAR TS. It is reasonable that the heatup could continue and that the restriction would be on criticality and power operation.

B. **Incorrect:** Heatup to normal operating temperature is not correct. The heatup can continue up to just below < 300 °F. The applicable TS 3.6.2.1 is applicable in MODES 1, 2, and 3*. Therefore the TS is not applicable until temperature is greater than or equal to 300 °F. **Plausible**: Many Technical Specifications are applicable in MODES 1 – 4. So it is reasonable that the CAR TS would be MODES 1 – 4. And that the TS could allow entry into a higher MODE with the equipment Inoperable. The action is plausible in that it is similar in timing (48 hours) for having one containment spray train and one containment cooling train inoperable.

C. **CORRECT:** Startup can continue and temperature can be raised to just below 300 °F (MODE 3). Entry to MODE 3 is not allowed as defined in TS 3.0.4. TS 3.6.2.1 is the appropriate TS and is applicable in MODES 1, 2 & 3*. Therefore it is allowed to stay in the current MODE; MODE 4.

D. **Incorrect**: Restore to OPERABLE within 48 hours or be in COLD SHUTDOWN within the next 36 hours is not correct since TS 3.6.2.1 is applicable in MODES 1, 2, and 3*, not MODES 1-4. **Plausible:** Many Technical Specifications are applicable in MODES 1 – 4 and that the TS could allow entry into a higher MODE with the equipment Inoperable. So it is reasonable that the CAR TS would be MODES 1 – 4. The action is plausible in that it is similar in timing (48 hours) for having one containment spray train and one containment cooling train inoperable.

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and making a decision based on knowledge of the Technical Specification and their bases. The candidate must apply Technical Specifications specific to CAR coolers (TS 3.6.2.1) and also assess whether heatup can continue based on the MODE the plant is in and TS 3.0.4.

References: TS 3.6.2.1, TS 3.6.2.1 Bases, TS 3.0.4, TS Table 1.1 - TS Rev through change 397,

Student Ref: NONE

Learning Objective: 282386 (01720-MB) Given a Limiting Condition for Operation, Applicability, and Action, and the Operability status of the associated systems, components, trains, or devices, determine if an Operational Mode change may be made.

280978 (02231-MB) Given any operating condition for the Containment Air Recirculation and Cooling Systems, state whether the condition requires entry into the Technical Specifications.

Question Source: New MP2 question. Question was written from a Calvert Cliffs question (106574) that was on their 2015 NRC SRO exam (Q13).

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

10CFR55: 43(b)(2) Facility operating limitations in the technical specifications and their bases

Comments (Question 90): 08/11/20 jwr. Completed changes to address the NRC's comments on 7/24/20. The NRC's comments were that the first part of answers "A" and "B", "Heatup can not continue" was not plausible and the "A" CAR fan should be specified in the second part of answers "B" and "D". A replacement statement for answers "A" and "B" was provided by the NRC and was used. It was "Heatup can continue to normal operating temperature". The "A" CAR fan was added to answers "B" and "D". The justification was also changed to support the answer changes and to clearly state why an answer was incorrect.

6/21/20 jwr. No answer was correct since the correct answer has HOT STANDBY instead of the correct HOT SHUTDOWN. Changed second part of answers to maintain current MODE or go COLD SHUTDOWN. Made question a lot easier to understand what is being asked. 6/4/20 jwr. Comments were received from both the first and second validation. This question was very confusing since both the first and second parts were all different. Changed question to make it a 2 X 2 to address validators comments. Changed A & C to maintain the plant in the current MODE DF 6/10/20
<< >> <<

>>QUESTION 91

The plant is operating at 100% power.

A number of alarms annunciate coincident with the following:

- Flow indicators HPSI FLOW TO LOOPS 1A (FI-311) and 1B (FI-321) are dark.
- Flow indicators LPSI FLOW TO LOOPS 1A (FI-312) and 1B (FI-322) are dark.
- The “A” SGFP Insert is dark.

1. What procedure will be entered to address this situation? AND

2. What actions are taken to mitigate the malfunction?

A. 1. AOP 2504A, Loss of Non-Vital Instrument Bus VR-11.
2. Take Local Manual control of FW-51A, #1 S/G Feed Regulating Valve.

B. 1. AOP 2504E, Loss of 120 VAC Vital Instrument Bus VA-30.
2. Take Local Manual control of FW-51A, #1 S/G Feed Regulating Valve.

C. 1. AOP 2504A, Loss of Non-Vital Instrument Bus VR-11.
2. Close Trip Circuit Breakers #3 & #7.

D. 1. AOP 2504E, Loss of Vital Instrument Bus VA-30.
2. Close Trip Circuit Breakers #3 & #7.

<<

>> QUESTION 91

K&A Rating: 016A2.02 (3.2)

K&A Statement: 016A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of power supply

Key Answer: **A**

Justification:

A. CORRECT: The indications provided indicate a loss of non-vital instrument bus VR-11. On a loss of VR-11, the #1 S/G Feed Reg Valve locks up. AOP 2504A, Loss of Non-Vital Instrument Bus VR-11, has the operator take local manual control of FW-51A, #1 S/G Feed Reg Valve.

B. **Incorrect:** A loss of VA-30 will affect the 'A' SGFP controls but not the Feed Reg Valve
Plausible: May confuse SGFP controls with Feed Reg Valve circuitry

C. **Incorrect:** A loss of VR-11 will not trip TCBs
Plausible: May think the results of a loss of VR-11 could cause a trip of TCBs after a period of time

D. **Incorrect**: A loss of VA-30 would require resetting TCBs #3 & #7.
Plausible: May think the indications given are caused by a loss of VA-30

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selection of an appropriate procedure during normal, abnormal, and emergency situations. The student is given a set of plant conditions that resulted from a loss of a Non-Nuclear Instrumentation power supply. And from the indications must select the correct procedure to correct, control and mitigate the consequences of the malfunction..

References: AOP 2504A, Loss of VR-11, 2504E, Loss of VA-30

Student Ref: NONE

Learning Objective: 282850 (05736-MB) Given a set of plant conditions, determine AOP 2504A/B/C/D/E/F, Loss of Non-Vital Instrument Panels VR-11 & VR-21 and Vital Instrument Panels VA-10, VA-20, and VA-30, & VA-40 applicability.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 91):

12AUG20 DF. Rewrote question based on comments made by NRC. The NRC's comment was that the question did not match the K/A.

5/20/20 jwr. This was one of the (10) questions sent to the NRC for their initial review. Their comment was that you could get the correct answer just by knowing system knowledge which makes it a RO question. They also noted that SRO questions should hit both of the items in the K/A. But if it can't hit both items it should be written to meet the second item in the K/A. They also suggested that a 2X2 would work and went thought what this question was revised to.<<

>>QUESTION 92

The plant experienced a large break LOCA 3.5 hours ago.

- Containment pressure is 6.89 psig and stable.
- MAIN STEAM LINE HI RAD/INST. FAIL (C-01, A-30) is NOT in alarm.
- HYDROGEN PURGE ISOL VLVS HI RAD (C-01, C-30) is IN alarm.

1. Procedurally, can Containment Spray be secured?

2. Why?

A. 1. No.
   2. CTMT pressure is greater than 4 psig.

B. 1. No.
   2. CTMT Hi range rad monitors ARE in alarm.

C. 1. Yes.
   2. Spray pumps have been operating for greater than 1 hour.

D. 1. Yes.
   2. Main Steam Line rad monitors ARE NOT in alarm.

<<

>>Question 92

K&A Rating: G2.4.45 (SRO 4.3)

K&A Statement: G2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm (027 Containment Iodine Removal)

Key Answer: **B**

Justification:

A. **Incorrect:** CTMT spray can terminated at less than 7 psig
Plausible: examinee may confuse 4 hours and 7 psig

B. **CORRECT:** CTMT spray cannot be terminated if CTMT high range RMs are alarming

C. **Incorrect:** CTMT spray needs to operate for at least 4 hours when actuated
Plausible: examinee may confuse 4 hours and 7 psig

D. **Incorrect**: MSL RMs have no tie to CTMT spray
Plausible: 'C' MSL RM can be used as B/U to CTMT hi range RMs

SRO Justification: the candidate needs to assess condition s and select an appropriate procedure during an abnormal situation. The question 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 mitigating strategy of a procedure.

References: EOP 2532 Rev. 35-00 step 14

Student Ref: NONE

Learning Objective: 283793/MB-05944 Describe time dependent actions and their bases for the following as specified in EOP 2532, Loss of Coolant Accident:

A. Termination of Containment Spray

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X

Comprehensive/Analysis:

10CFR55: 10CFR43(b)(5) SRO assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations

Comments:

>>QUESTION 93

The Facility 2 Control Room Air Conditioning (CRAC) system is operating in the “Normal” Mode and the Facility 1 CRAC system is not operating.

The fuse that supplies power to Radiation Monitor RM 9799A blows.

1. What is the status on the CRAC system as a result of this fuse failure? AND

2. What procedure will be used to address the condition of the CRAC system?

A. 1. Facility 1 Filter fan operating. Facility 1 Supply and Exhaust fans are not operating.
   2. AOP 2588, Loss of Cooling, Ventilation or Control Room Envelope Boundary Breach.

B. 1. Facility 1 Filter fan, Supply fan and Exhaust are operating.
   2. AOP 2588, Loss of Cooling, Ventilation or Control Room Envelope Boundary Breach.

C. 1. Facility 2 Filter fan is not operating. Facility 2 Supply and Exhaust fans are operating.
   2. ARP 2590-159, CRACS IN AUTO RECIRC. MODE.

D. 1. Facility 2 Filter fan, Supply fan and Exhaust are operating.
   2. ARP 2590-159, CRACS IN AUTO RECIRC. MODE.

<<

>>QUESTION 93

K&A Rating: 072A2.03 (2.9)

K&A Statement: 072A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the ARM system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Blown power-supply fuses.

Key Answer: **C**

Justification:

A. **Incorrect:** The first statement is correct that the Facility 1 Filter fan will be operating and the Facility 1 Supply and Exhaust fans will not be operating. AOP 2588, Loss of Cooling, Ventilation or Control Room Envelope Boundary Breach is also not correct because this procedure will not correct the system transferring to recirculation.
Plausible: The first statement is correct and going to an AOP for Control Room Air Conditioning is reasonable.

B. **Incorrect:** The Facility 1 Filter fan will be operating and the Facility 1 Supply and Exhaust fans will not be operating. AOP 2588, Loss of Cooling, Ventilation or Control Room Envelope Boundary Breach is also not correct. The loss of power to radiation monitor 9799A will only start the "A" the Facility 1 filter fan F32A not the non operating Facility 1 Supply and Exhaust fans.
Plausible: Most systems when they get a safety signal will actuate all required equipment on that facility. CRAC is not like most safety systems, in that it requires Operator action to ensure complete actuation. Going to AOP 2588, Loss of Cooling, Ventilation or Control Room Envelope Boundary Breach for a problem with Control Room Air Conditioning system is reasonable.

C. **CORRECT:** The fuse failure will result in a Facility 1 Recirculation signal which will start the Facility 1 Filter fan and isolate outside air. It will not start the non operating Facility 1 Supply and Exhaust fans. ARP 2590A-159 CRACS IN AUTO RECIRC MODE will alarm and is the correct procedure to address this condition. The CRAC radiation monitor is an area radiation monitor and is identified as such in TS

D. **Incorrect**: The fuse failure will result in a Facility 1 Recirculation signal which will start the Facility 1 Filter fan but will not start the non operating Facility 1 Supply and Exhaust fans.
Plausible: It is reasonable that an actuation of one of the radiation monitors would result in both facilities of CRAC shifting to recirculation. ARP 2590A-159 CRACS IN AUTO RECIRC MODE will alarm and is the correct procedure to address this condition.

<u>SRO Only Justification</u>: This question is SRO only as it requires assessing plant conditions and then selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must understand the condition of the CRAC system after the loss of one of the CRAC intake duct radiation monitor and then select the correct procedure to mitigate the affect of the failure.

References: ARP 2590A-159 Rev 000-04, AOP 2588 Rev 001, 25203-28500 SH. 660 Rev 4 25203-32023 SH 5 Rev 19, 25203-32023 SH 6 Rev 18, CRA-00-C Rev 4

Student Ref: NONE

Learning Objective: 281109 (02301-MB) As given in CRA-00-C, describe the effects on the Control Room Air Conditioning System of a loss or malfunction of the following: D) Radiation Elements RE 9799A and /or 9799B.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 93): 08/10/2020 jwr. Completed change to add the NRC's recommendation from their 7/24/20 review. The NRC discussed whether the CRAC RM was an area radiation monitor. It is an area radiation monitor as delineated in TS. The NRC requested we put that it is an area radiation monitor into the justification. The justification for the correct answer was updated with this information.

6/2/20 jwr. Comments from validators related to the normal operating OP 2315A being the correct procedure to go to being incorrect. Since OP 2315A does have a section to transfer to normal from recirc and from normal to recirc. The validators felt that OP 2315A could be a correct procedure to go to, just not the best procedure. Selected another procedure, AOP 2588 Loss of Cooling, Ventilation or Control Room Envelope Boundary Breach, as an incorrect procedure.
<<>QUESTION 94

The plant is operating at 100% power with the "B" Auxiliary Feedwater (AFW) Pump out of service for maintenance.

Then the following events occur:

- The plant automatically trips due to a Steam Generator Tube Rupture (SGTR) on #2 Steam Generator (SG).
- The RSST <u>and</u> VA-20 are lost at the time of trip.
- A Safety Injection Actuation Signal (SIAS) is automatically actuated.

Which one of the following actions must the US direct during EOP 2525, Standard Post Trip Actions, to mitigate the consequences of this event?

A. Dispatch an Operator to the Hot Shutdown Panel, C-21, to throttle open the #2 Atmospheric Dump Valve.

B. Direct the BOP to swap the control power supply switch for the Terry Turbine to Facility 1.

C. Dispatch a PEO to manually operate the "B" Auxiliary Feedwater Regulating Valve, 2-FW-43B.

D. Direct the BOP to close the associated Disconnect, and then close the #2 SG Steam Supply to the Terry Turbine, MS-202.

<<

>>QUESTION 94

K&A Rating: Generic Conduct of Operations 2.1.30 (4.0)

K&A Statement: 2.1.30 Ability to locate and operate components, including local controls.

Key Answer: **C**

Justification:

A. **Incorrect:** A loss of VA-20 will result in a loss of power to the #2 ADV from ALL remote locations. The #2 ADV can ONLY be operated locally with the handwheel in manual.
Plausible: The examinee may think that the C-21 components have a different power supply than they have on the main control boards.

B. **Incorrect:** Control power to the Turbine Driven Auxiliary Feedwater Pump (TDAFP) is from DV-20, NOT VA-20; therefore, swapping power supplies will have NO impact on the availability of the TDAFP.
Plausible: The examinee may not remember that the power supply for the TDAFP is DV-20 NOT VA-20.

C. **CORRECT:** On a loss of normal power, Condensate is lost; therefore, Main Feedwater is lost. The loss of VA-20 will cause the "B" Auxiliary Feed Regulating Valve to fail open. EOP 2525 requires two Auxiliary Feed Pumps operating. In order to prevent overfeeding the #2 SG, the "B" Auxiliary Feed Regulating Valve, 2-FW-43B, must be manually controlled or isolated locally.

D. **Incorrect**: The #2 SG Steam Supply to the Terry Turbine, MS-202, is not closed in EOP 2525. There is no step in EOP 2525 to perform this. Others EOP steps should not be performed when in EOP 2525. This action is only performed in EOP 2534, after lowering both hot leg temperatures to <515°F, when isolating the affected S/G.
Plausible: Because this action is performed in EOP 2534, after transitioning out of EOP 2525, as part of SG isolation.

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and then selection of appropriate procedures during normal, abnormal, and emergency situations.

References: EOP 2525 Rev. 028-00, OP-2260 Rev. 015-00, AOP-2501 Rev. 003

Student Ref: NONE

Learning Objective: 283646 (00381-MB) Evaluate the emergency condition and determine the appropriate course of action in accordance with EOP 2525, Standard Post Trip Actions.

Question Source: 451807 (9000005-MB) NRC 2009 Q4 Retake U-SRO

Question History: 451807 (9000005-MB) NRC 2009 Q4 Retake U-SRO.
Edited to shorten and make more accurate and readable.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 94): 08/12/20 jwr. Completed changes in response to the NRC's 7/24/20 comments. The NRC commented that the second part of the question was not needed and should be removed. Remove "and what is the reason for this action" from the stem and removed second part of all the answers.
<<

>>QUESTION 95

The Charging pumps and High Pressure Safety Injection (HPSI) pumps can be made not capable of injecting but available at short notice to balance Low Pressure Overpressure (LTOP) requirements and Shutdown Risk.

Which of the following configurations are procedurally allowed to make the Charging and HPSI pumps not capable of injecting?

1. How are the Charging pumps made not capable of injecting? AND

2. How are the HPSI pumps made not capable of injecting?

A. 1. Shutting the Charging discharge valve on C-02/3.
   2. Shutting the HPSI injection valves on C-01.

B. 1. Shutting the Charging discharge valve on C-02/3.
   2. Placing the HPSI pump control switch in PTL and removing the breaker control power fuses.

C. 1. Placing the Charging pump control switch in the Pull-To-Lock (PTL) position.
   2. Placing the HPSI pump control switch in PTL and removing the breaker control power fuses.

D. 1. Placing the Charging pump control switch in the Pull-To-Lock (PTL) position.
   2. Shutting the HPSI injection valves on C-01.

<<

>>QUESTION 95

K&A Rating: Generic Equipment Control 2.2.14 (4.3)

K&A Statement: 2.2.14 Knowledge of the process for controlling equipment configuration or status

Key Answer: **C**

Justification:

A. **Incorrect:** Shutting the Charging discharge valve on C-02/3 is not a configuration in the TS bases for making the Charging pumps not capable of injecting. Shutting the HPSI injection valves on C-01 is also not correct.
Plausible: Shutting the Charging discharge valve on C-02/3 is similar to the requirement of shutting the HPSI discharge valve(s) on C-01.

B. **Incorrect**: Shutting the Charging discharge valve on C-02/3 is not a configuration in the TS bases for making the Charging pumps not capable of injecting.
Plausible: Shutting the Charging discharge valve on C-02/3 is similar to the requirement of shutting the HPSI discharge valve(s) on C-01. And Placing the HPSI pump control switch in PTL and removing the breaker control power fuses is correct

C. **CORRECT**: Placing the Charging pump control switch in the Pull-To-Lock (PTL) position and placing the HPSI pump control switch in PTL and removing the breaker control power fuses is correct. These are procedure allowed methods.

D. **Incorrect:** Shutting the HPSI injection valves on C-01 is not correct..
Plausible: Placing the Charging pump control switch in the Pull-To-Lock (PTL) position is correct and shutting the HPSI injection valves is similar to the requirement of shutting the HPSI header discharge valve(s).

<u>SRO Only Justification</u>: This question is SRO only as it requires the examinee to know Facility operating limitations in the Technical Specifications and their bases.

References: Technically Specifications Bases 3.4.9.3, SP 2619A-003 Rev.034
Student Ref: NONE

Learning Objective: 216576 Explain the bases for all Technical Specifications that apply to plant cooldown.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 43.3

Comments (Question 95): 08/10/2020 jwr. Completed NRC recommended change from their review on 7/24/20. They recommended that the second line of the stem be changed to remove “provided in the Technical Specification Bases” and replaced with “procedurally allowed”. Also added to the correct answer justification that these are procedurally allowed methods.

6/1/20 jwr. Made changes based on PF comments to change control boards to panel numbers. Also made change as suggested to change HPSI discharge valves to injection valves because the tagging requirement would be somewhat vague since we normally tag all of these components when we make components not capable of injecting. 5/25/20 jwr. This question was replaced with a new question. The initial OPS validation was that this question is really something very specific and something an Operator would review in a procedure since it is not done often. The new question is of high importance, deals with configuration control, is covered in the TS bases and is something performed each cooldown and heatup.
<<

>>QUESTION 96

The plant is starting up from a mid cycle shutdown. The following are the present plant conditions:

- Both Shutdown Group CEAs are fully withdrawn.
- Withdrawal of Regulating Group CEAs has not commenced.
- RCS temperature is 532 °F.

The Unit Supervisor (US) declares all (3) Auxiliary Feedwater pumps Inoperable when the Shift Manager is notified that the bearing oil was replaced in all three pumps with the wrong oil.

What MODE is the Unit in and what action will the US direct in accordance with Technical Specifications?

A. MODE 2.
Direct actions to stay in current MODE.

B. MODE 2.
Direct actions to bring the Unit to MODE 4.

C. MODE 3.
Direct actions to stay in current MODE.

D. MODE 3.
Direct actions to bring the Unit to MODE 4.

<<

>>QUESTION 96

K&A Rating: Generic Equipment Control 2.2.35 (4.5)

K&A Statement: 2.2.35 Ability to determine Technical Specification Mode of Operation.

Key Answer: **C**

Justification:

A. **Incorrect:** MODE 2 is not correct. The Reactor Startup procedure specifies MODE 2 when Regulating Group 4 CEAs have been withdrawn to 72 steps. Regulating Group CEA has not begun.
Plausible: The examinee must know when MODE 2 is entered to differentiate between MODE 2 and 3. Direct action to stay in the current MODE is correct.

B. **Incorrect:** MODE 2 is not correct. The Reactor Startup procedure specifies MODE 2 when Regulating Group 4 CEAs have been withdrawn to 72 steps. Regulating Group CEA has not begun.
Plausible: Direct action to bring the unit to MODE 4 is plausible. This is what is required if two AFW pumps are Inoperable.

C. **CORRECT:** The plan is in MODE 3 (< 0.99 K_{EFF} and > or = to 300 °F). The Reactor Startup procedure specifies MODE 2 when Regulating Group 4 CEAs have been withdrawn to 72 steps. Technical Specification 3.7.1.2 directs, with three Auxiliary Feedwater pumps Inoperable in MODE 1, 2, or 3, that a MODE change not be made. The specified required action is "LCO 3.0.3 and other LCO required ACTIONS requiring MODE changes are suspended until one AFW pump is restored to OPERABLE status."

D. **Incorrect:** Direct action to bring the unit to MODE 4 is not correct. TS 3.7.1.2 directs, with three AFW pumps Inoperable that a MODE change not be made.
Plausible: Direct action to bring the unit to MODE 4 is plausible. This is what is required if two AFW pumps are Inoperable.

<u>SRO Only Justification</u>: This question is SRO only as it pertains to Conditions and limitations in the facility license. The examinee must understand the MODE the plant is in, actions required in Technical Specifications, and MODE applicability for the AFW pumps TS.

References: TS 3.7.1.2 and TS Table 1.1, OP 2202 rev. 025 Student Ref: None

Learning Objective: 282385 (01719-MB) Define each of Operational Modes 1 through 6 in terms of mode title, K-eff, percent thermal power, and average reactor coolant temperature.
282386 (01720-MB) Given a Limiting Condition for Operation, Applicability, and Action, and the Operability status of the associated systems, components, trains, or devices, determine if an Operational Mode change may be made.

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 43(b)(1) Conditions and limitations in the facility license.

Comments (Question 96): 08/16/2020 jwr. Replaced question in response to the NRC 07/24/20 comments. Their comments were that it was essentially the same as Q90 because it involved TS 3.0.4. Their suggestion was to maybe make the question a different MODE and where TS 3.0.4 isn't applicable. Wrote a new question were TS 3.0.4 did not apply and in a different MODE.

5/20/20 jwr. This question was replaced with an industry bank question with some modification. This was one of the (10) questions sent to the NRC for their initial review. Comments included that this does not determine the TS Mode of operation. The suggestion was to put temperatures in the answers instead of MODE. Also commented that we should use "train" or "Facility" instead of "ECCS subsystem" unless that is what it says in the TS. Note the TS uses "subsystem".
<<

>>QUESTION 97

The plant is shutdown for a refueling outage.

- The plant entered MODE 3 on April 1 at 2300
- Fuel off load is ready to commence on April 7 at 1900.

1. Can fuel movement commence? AND

2. What is the Technical Specification Basis?

A. 1. NO
   2. Heat from off-loaded fuel <u>will</u> exceed Spent Fuel Pool Cooling capacity.

B. 1. NO
   2. Short-lived fission products <u>have not</u> decayed enough such that the radiological dose consequences of a fuel handling accident are bounded.

C. 1. YES
   2. Heat from off-loaded fuel <u>will not</u> exceed Spent Fuel Pool Cooling capacity.

D. 1. YES
   2. Short-lived fission products <u>have</u> decayed enough such that the radiological dose consequences of a fuel handling accident are bounded. <<

>>QUESTION 97

K&A Rating: 2.3.14 (3.8)

K&A Statement: 2.3.14 Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.

Key Answer: **D**

Justification:

A. **Incorrect:** Fuel movement can not commence is not correct. The time since subcriticality is > 100 hours (April 1 at 2300 to April 7 at 1900 is 140 hours). And heat from the off loaded fuel will exceed Spent Fuel Pool Cooling capacity is also not correct
Plausible: The TS time requirement has been changed from 150 hours to 100 hours. And the basis for the TRM requirement to maintain the reactor in MODE 5 or 6, or defueled for 616 hours is SFPC capacity.

B. **Incorrect:** Fuel movement can not commence is not correct. The time since subcriticality is > 100 hours (April 1 at 2300 to April 7 at 1900 is 140 hours) so fuel movement can commence.
Plausible: The TS time requirement has been changed from 150 hours to 100 hours.

C. **Incorrect**: Heat from the off loaded fuel will not exceed Spent Fuel Pool Cooling capacity is not correct.
Plausible: Decay heat does lower as a function of time from reactor shutdown. The basis for the TRM requirement to maintain the reactor in MODE 5 or 6, or defueled for 616 hours is SFPC capacity.

D. **CORRECT**: Fuel movement can commence and the basis of decay of short-lived fission products so that the radiological dose consequences of a fuel handling accident are bounded.is correct. TS 3.9.3.1 specifies that the reactor shall be subcritical for a minimum of 100 hours (April 1 at 2300 to April 7 at 1900 is 140 hours) prior to movement of irradiated fuel in the reactor pressure vessel. The basis states that the minimum requirement for reactor subcriticality prior to movement of irradiated fuel ensures that sufficient time has elapsed to allow the radioactive decay of short-lived fission products such that the calculated radiological dose consequences of the fuel handling accident are bounding.

SRO Only Justification: This question is SRO only as it pertains to radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. This question requires Technical Specification LCO and basis knowledge of radiation hazards associated a fuel handling accident which is an abnormal condition.

References: TS 3.9.3.1 Refueling Operations Decay Time Student Ref: NONE

Learning Objective:

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments (Question 97): 09/09/2020 jwr. Wrote new question to address Don Jackson's comments from 09/02/2020. Don's comment was that this was not and SRO level question. The suggestion was to replace this question.
<<

>>QUESTION 98

The plant experienced an emergency event and the SERO has been fully activated. A PEO must perform a task in the MP2 Auxiliary Building that is estimated to require 10 REM TEDE exposure.

What SERO position is authorized to raise the exposure limits of this PEO?

A. Manager of Radiological Dose Assessment (MRDA).

B. Assistant Director, Technical Support (ADTS).

C. Manager of Radiological Consequence Assessment (MRCA).

D. Manager of Control Room Operations (MCRO).

<<

>> QUESTION 98

K&A Rating: Generic, Radiation Control 2.3.4 (3.7)

K&A Statement: 2.3.4 Knowledge of Radiation exposure limits under normal or emergency conditions.

Key Answer: **B**

Justification:

A. **Incorrect:** Manager of Radiological Dose Assessment (MRDA).is not correct. .
Plausible: The MRDA recommends emergency exposure upgrades for the SERO.

B. **CORRECT**: The Assistance Director, Technical Support (ADTS) is correct. FAP09, Radiation Exposure Controls, Attachment 2, Responsibilities and Attachment 3, Emergency Exposure Control Guidance clearly delineate that the ADTS is authorized to issue emergency exposure upgrades for SERO emergency workers outside the protected area fence.

C. **Incorrect:** Manager of Radiological Consequence Assessment (MRCA) is not correct.
Plausible: The MRCA recommends emergency exposure upgrades for on-site SERO.

D. **Incorrect**: The Manager of Control Room Operations (MCRO) is not correct.
Plausible: The MCRO is the control room shift manager. This position has the full responsibility of the DSEO, including exposure upgrades, until DSEO responsibilities are assumed by the EOF DSEO upon activation of the Emergency Organization.

SRO Only Justification: This question is SRO only as it requires assessing plant conditions and applying knowledge of Emergency Plan procedure requirements. This is SRO knowledge in that many procedures, both emergency operating procedures and emergency plan procedures, are in affect and it is the SROs responsibility to comply with the requirements of all of them as he assumes his various roles and duties.

References: Millstone Station Emergency Plan Rev. 60, MP-26-EPI-FAP09 Rev. 006, MP-26-EPI-FAP09-001 Rev. 000

Student Ref: NONE

Learning Objective: 205205 (00216-EP) Explain the requirements for exposure control during a radiological incident.

Question Source: Bank

Question History: New but written from Q98 from 2000 NRC exam. VISION 451470 (1000018-MB). Two of the incorrect answers changed and stem significantly modified.

Cognitive Level: Memory/Fundamental Knowledge: X
Comprehensive/Analysis:

10CFR55: 43(b)(4) Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments (Question 98): 08/10/2020 jwr. Completed changes recommended by the NRC on 7/24/202. The NRC recommended that answer “D” be replaced with the Manager of Control Room Operations (MCRO). This will make it more relevant to the SRO position.
<<

>>QUESTION 99

The plant was operating at 100% power when a Steam Generator Tube Leak (SGTL), in excess of Technical Specification limits, occurred and worsened.

A manual reactor trip was attempted when the Reactor Coolant System leak exceeded 150 gpm.

The Manual Trip pushbuttons did not trip the reactor. The reactor was successfully tripped when the CEDM output breakers were opened at C04.

The following conditions exist 10 minutes post trip:

- SIAS, CIAS, and EBFAS equipment has fully actuated on Facility 1.
- A Main Steam Safety Valve is stuck open, below its blowdown setting, on the affected Steam Generator.

Given the EAL tables, what is the correct Classification and its' Major Heading?

A. Alert, Barrier Failure.

B. Alert, Equipment Failure.

C. Site Area Emergency, Barrier Failure.

D. Site Area Emergency, Equipment Failure.

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>>QUESTION 99

K&A Rating: Generic, Emergency Procedures/Plan 2.4.41 (4.6)

K&A Statement: G2.4.41 Knowledge of the emergency action level thresholds and classifications.

Key Answer: **C**

Justification:

A. **Incorrect:** Alert is not correct because two barriers are affected; RCB4 (P) and CNB3 (L), which makes the classification a SAE.
Plausible: The candidate could reasonably make the case that because RCB4 is a potential loss it should not be used for classification.

B. **Incorrect:** Alert is not correct since it is a SAE as a result of two barriers.
Plausible: The candidate could reason that it is an Alert because the reactor did not trip when the manual trip buttons were pressed but manual trip was successful because the CEDM breakers were opened. But there was no automatic reactor trip failure. Therefore this classification is not correct.

C. **CORRECT:** The classification is a SAE based on two barriers affected. RCB4 (P) because RCS leakage (> 150 gpm) exceeds CVCS capacity and EOP 2525 was entered and CNB3 (L) because there is primary to secondary > Tech Spec limits and there is a non isolable steam release from the affected S/G to the environment. And from the barrier reference table two barriers either (L) or (P) results in a Site Area Emergency.

D. **Incorrect:** The SAE under Equipment Failure is not correct since a manual reactor trip on C04 includes both the manual pushbuttons and CEDM breakers. In additional the SAE on barrier failure would be the correct classification even with an SAE on equipment failure. A single classification must be made top to bottom and left to right on the tables.
Plausible: It is reasonable that the candidate believe that a manual reactor trip attempted at panel C04 includes just the manual reactor trip pushbutton and not the CEDM breakers and since reactivity is the highest safety function that it should be classified above the barrier failure.

<u>SRO Only Justification</u>: This question is SRO only as it requires assessing plant conditions and then selection of appropriate procedures during normal, abnormal, and emergency situations. Classification is also solely an SRO responsibility.

References: MP-26-EPI-FAP06-002 Millstone Unit 2 Emergency Action Levels
Student Ref: MP-26-EPI-FAP06-002 Millstone Unit 2 Emergency Action Levels

Learning Objective: 283632 (MB-06651) Using EAL Tables in FAP06, EAL Basis Document and given plant conditions, classify an event with 100% accuracy.

Question Source: New

Question History: New.

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 99): 08/10/2020 jwr. Completed the NRC's recommended changes from their 7/24/2020 review. They recommended the following changes to the stem; remove the extra space before 100%, replace CVCS capacity with a value, and replace faulted with whatever termilogy we use. The space was removed, "CVCS capacity" was replaced with "150 gpm", and "faulted" was replaced with "affected".

5/18/2020 jwr shortened stem by removing 3rd bullet on carrying out 2525. OPS validation comment.
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>>QUESTION 100

The plant is at 100% power with the following conditions;

- AOP 2560, “Storms Winds, and High Tides” has been entered.
- Condenser back pressure is 4.5 in Hg and getting worse.
- AOP 2574, “Loss of Condenser Vacuum” has been entered.
- Circulating Water pump speeds are at 60%.
- Traveling Screen differential pressures are 12 inches and rising.

What procedure will be used <u>and</u> what action will be taken to address the loss of vacuum condition?

A. The US will <u>enter</u> AOP 2575, "Rapid Downpower” and lower power to restore Condenser back pressure.

B. The US will <u>exit</u> AOP 2574 and enter AOP 2575, "Rapid Downpower”, and lower power to restore Condenser back pressure.

C. The US will <u>enter</u> AOP 2517, "Circulating Water Malfunctions” and raise Circulating Water pump speeds to lower Condenser back pressure.

D. The US will <u>exit</u> 2574 and enter AOP 2517, "Circulating Water Malfunctions”, and raise Circulating Water pump speeds to lower Condenser back pressure.

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QUESTION 100

K&A Rating: Generic, Emergency Procedures/Plan 2.4.11 (4.2)

K&A Statement: 2.4.11 Knowledge of abnormal condition procedures.

Key Answer: **A**

Justification:

A. **Correct**: The US will enter AOP 2575, "Rapid Downpower" and lower power to restore Condenser back pressure. AOP 2574 is not exited.

B. **Incorrect:** The US will not exit AOP 2574, "Loss of Condenser Vacuum".
Plausible: The examinee could reason that going to AOP 2575, Rapid Downpower" will address the problem so there is no need to stay in AOP 2574, "Loss of Condenser Vacuum".

C. **Incorrect**: The US will enter AOP 2517, "Circulating Water Malfunctions" and raise Circulating Water pump speeds to lower Condenser back pressure is not correct. When Traveling Screen DPs are 12 inches and above procedures do not direct raising speeds.
Plausible: The AOP 2574 does have a contingency action to refer to AOP 2517 and raising Circulating Water pump speeds will increase heat transfer and lower Condenser backpressure.

D. **Incorrect:** The US will not exit AOP 2574, "Loss of Condenser Vacuum" nor raise Circulating Water pump speeds. When Traveling Screen DPs are 12 inches and above procedures do not direct raising speeds.".
Plausible: The AOP 2574 does have a contingency action to refer to AOP 2517 and raising Circulating Water pump speeds will increase heat transfer and lower Condenser backpressure

SRO Only Justification: This question is SRO only as it requires assessment of facility conditions and selection of appropriate procedures during normal, abnormal and emergency situations.

References: AOP 2574 rev. 009, AOP 2517 rev. 008, OP 2255 rev. 003
Student Ref: None

Learning Objective: 283397(05567-MB) Outline the major actions for AOP 2574, "Loss of Condenser Vacuum".

Question Source: New

Question History: New

Cognitive Level: Memory/Fundamental Knowledge:
Comprehensive/Analysis: X

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

Comments (Question 100): 08/24/20 jwr. A new question was written in response to the NRC's feedback on 08/20/20. Their feedback was that it is a AOP K/A. Even though it is under the "Emergency Procedures / Plan" heading it should be an AOP question. Wrote new AOP question.

08/13/20 jwr. The question was replaced in response to the NRC's comments on 7/24/20. The NRC's comment was that the question was not at the SRO level. They provided general guidance for an SRO question like this is to "enter a procedure and specify a major action". This question, due to the procedures involved, could not be made into an acceptable question.

6/5/20 jwr - wrote a new question because a third plausible answer could not be written and because the other answers were so close. 4/30/20 jwr - Comments from Gerry Baker changed answer C to be more precise G. Baker feedback 2/24/20:
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>>QUESTION 21

How does AOP 2577, "Fuel Handling Accident", direct ventilation be aligned in response to a fuel handling accident, when containment purge is in service?

A. Push CPVIS actuation pushbuttons (C01), and
Place the Control Room Air Condition System in "RECIRC".

B. Push CPVIS actuation pushbuttons (C01), and
Place the Enclosure Building Filtration system on the Enclosure Building.

C. Close the Containment Purge Dampers 2-AC-~~5~~, ~~6~~, ~~7~~ & ~~8~~ 4, 5, 6 & 7, and
Place the Control Room Air Condition System in "RECIRC".

D. Close the containment purge dampers 2-AC-~~5~~, ~~6~~, ~~7~~ & ~~8~~ 4, 5, 6 & 7, and
Place the Enclosure Building Filtration system on the Enclosure Building.

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>>QUESTION 64

The plant is operating at 100% power. Containment venting (depressurization) is in progress.

1. What flow path is being used? AND
2. What radiation monitor will isolate the flow path if a high radiation condition in Containment occurs?

A. 1. Hydrogen Purge valves EB-99 & 100 or EB-91 & 92.
   2. Containment High Range Radiation Monitors RM-8240 or RM-8241.

B. 1. Hydrogen Purge valves EB-99 & 100 or EB-91 & 92.
   2. Containment Radiation Monitors RM-8123A/B or RM-8262A/B.

C. 1. Containment Purge Isolation Dampers AC-~~5~~, ~~6~~, ~~7~~ & ~~8~~. 4, 5, 6 & 7
   2. Containment High Range Radiation Monitors RM-8240 or RM-8241.

D. 1. Containment Purge Isolation Dampers AC-~~5~~, ~~6~~, ~~7~~ & ~~8~~. 4, 5, 6 & 7
   2. Containment Radiation Monitors RM-8123A/B or RM-8262A/B.

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>>QUESTION 95

The Charging pumps and High Pressure Safety Injection (HPSI) pumps can be made not capable of injecting but available at short notice to balance Low ~~Pressure~~ Temperature Overpressure (LTOP) requirements and Shutdown Risk.

Which of the following configurations are procedurally allowed to make the Charging and HPSI pumps not capable of injecting?

1. How are the Charging pumps made not capable of injecting? AND

2. How are the HPSI pumps made not capable of injecting?

A. 1. Shutting the Charging discharge valve on C-02/3.
   2. Shutting the HPSI injection valves on C-01.

B. 1. Shutting the Charging discharge valve on C-02/3.
   2. Placing the HPSI pump control switch in PTL and removing the breaker control power fuses.

C. 1. Placing the Charging pump control switch in the Pull-To-Lock (PTL) position.
   2. Placing the HPSI pump control switch in PTL and removing the breaker control power fuses.

D. 1. Placing the Charging pump control switch in the Pull-To-Lock (PTL) position.
   2. Shutting the HPSI injection valves on C-01.

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