| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 1 | Tier # | 2 | 2 |
| K/A Statement: PRT: Predict impact / mitigate | Group # | 1 | 1 |
| exceeding PRT high pressure limits | K/A # | 007A2.05 | |
| Proposed Question: | Importance Rating | 3.2 | 3.6 |

With the plant at 100% power, a relief valve sticks partially open, resulting in the following sequence of events:

1. The PZR REL TK PRESSURE HI annunciator is received on MB4.
2. The relief valve reseats.
3. The RO reports PRT pressure has increased to 65 psia and is now stable.
4. The US directs the RO to lower PRT pressure per OP 3301A, *Pressurizer Relief Tank and Reactor Vessel Flange Leakoff Operations*, Section 4.7, "Restoring PRT Pressure to Normal".
5. The RO ensures Ctmt Isolation Valves related to the PRT are in the proper positions at MB1.

Complete the following statements.

Reaching 65 psia exceeded the pressure where Rupture Disk ____(1)____occur. Per OP 3301A, the first action(s) the RO will take to restore PRT pressure to normal is to ____(2)____at MB4.

a) (1) fatigue or distortion may
(2) press the BLOCK AUTO AV8031 pushbutton and open PRT Drain Valve 3DGS-AV8031
b) (1) fatigue or distortion may
(2) open PRT Vent Valve 3RCS-PCV469
c) (1) rupture was expected to
(2) press the BLOCK AUTO AV8031 pushbutton and open PRT Drain Valve 3DGS-AV8031
d) (1) rupture was expected to
(2) open PRT Vent Valve 3RCS-PCV469

Proposed Answer: A

Explanation (Optional):
The PRT receives hot water or steam from the relief valve that discharge to it. This raises PRT level, and pressure as level increases and the water in the PRT heats up. PRT pressure has reached 65 psia, which is above the design discharge pressure of 64.7 psia where Rupture Disk fatigue or distortion may occur.
"C" and "D" are wrong, since PRT pressure is below the pressure at which the PRT Rupture Disks are designed to rupture, which is 91 psig (105.7 psia). "C" and "D" are plausible, since PRT pressure is well above the high pressure alarm setpoint of 23 psia, and the Rupture Disks' design pressure limit has been exceeded, and PRT pressure where rupture is expected to occur is actually a range from 86 to 100 psig (100.7 to 114.7 psia).
"A" is correct, since the method used to lower PRT pressure is to open its drain valve, and then pump down the PRT. This will be followed by reopening the PRT vent valve and refilling it with cold water. The first action the RO will at MB4 is to press the BLOCK AUTO AV8031 pushbutton and OPEN the PRT Drain Valve (3DGS-AV8031). Pushing the "BLOCK AUTO SIGNAL" pushbutton is required since the PRT Drain Valve will not open with PRT level below its Hi Level setpoint of 86% and it will automatically close if level drops below the High Level setpoint if the pushbutton is not depressed.
"B" is wrong, since the PRT High Pressure annunciator is lit, and the PRT vent valve automatically closes on a PRT high pressure condition. "B" is plausible, since PRT pressure is high, and opening the vent valve on a tank with high pressure will lower tank pressure. Also, the RO will open the vent valve from MB4 as part of this procedure section after pressure has been returned to normal.

| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided, including version/revision number.) | OP 3301A (Rev. 11), Precautions 3.1 and 3.6 |
| | OP 3301A (Rev. 11), Section 4.7 |
| | OP 3353.MB4A (Rev. 07), 2-4 |
| | P&ID 102F (Rev. 17), Note 3 |
| | P&ID 107A (Rev. 28) |

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Pressurizer Relief Tank System, including the basis for each

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.3, 41.7, 41.10, and 41.14

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 2 | Tier # | 1 | 1 |
| K/A Statement: Pzr Level Control Malfunction: | Group # | 2 | 2 |
| Reasons for false indication of Pzr level when PORV or Spray Valve is open and the RCS is saturated | K/A # | APE028.AK3.03 | |
| | Importance Rating | 3.5 | 4.1 |

Proposed Question:

With the plant initially at 100% power, the following sequence of events occurs:

1. The “B” Pzr PORV fails open.
2. The crew is NOT able to close the “B” PORV Block Valve.
3. Safety injection actuates.

Fifteen minutes later, conditions are as follows:

- RCS pressure is 1600 psia.
- Subcooling based on core exit thermocouples is 0°F.

What does Pzr level currently indicate on MB4, and why?

a) Empty, since RCS inventory initially decreased, lowering RCS pressure to the point where mass injected from the RWST equals mass lost out the PORV.
b) Empty, since RCS pressure has stabilized above where ECCS flow can make up for mass loss out the PORV. This is due to decay heat being greater than heat being removed out the break.
c) Full, since saturation conditions exist. This has caused a two-phased mixture to form in the vessel and hot legs, which has expanded RCS inventory up the surge line into the Pressurizer.
d) Full, since RCS mass being lost out the PORV is less than mass being injected from the RWST, so overall RCS inventory has been increasing.

Proposed Answer: C

Explanation (Optional):
“C” is correct, and “A” and “B” wrong, since a PORV failing open will cause pressure to decrease as energy is removed from the Pzr. This causes Pzr steam space pressure to decrease, causing saturated water to flash to steam, removing heat from the RCS. As pressure drops, saturation will be reached in the reactor vessel and RCS hot legs, causing a two-phase mixture to form. This causes RCS inventory to become less dense, expanding it into the Pzr, raising Pzr level.
“A” is plausible, since Pressurizer level normally experiences the rapid decrease on a Reactor Trip as RCS cools down, and continues to decrease on a LOCA as mass lost out the break exceeds ECCS injection until RCS pressure drops to where equilibrium is reached. This is seen for LOCAs in other locations, such as a hot leg break or a SGTR.
“B” is plausible, since Pressurizer level normally experiences the rapid decrease on a Reactor Trip as, and continues to decrease on a LOCA as mass lost out the break exceeds ECCS injection. For a small Cold Leg break, pressure stabilizes with mass being lost out the break exceeding ECCS injection flow due to the reactor vessel acting as a Pressurizer, with a steam bubble in the head, and the fuel acting as heaters.
“D” is wrong, since RCS pressure has stabilized below initial RCS pressure. If RCS inventory were increasing with a full Pressurizer, RCS pressure would be increasing.
“D is plausible, since on breaks of very small size, SIS will actuate, and when ECCS starts injecting, mass into the RCS exceeds mass lost out the break. But in this case, RCS pressure increases to the PORV setpoint, and the PORVs release excess RCS inventory.

Technical Reference(s): Millstone MCORE07 PowerPoint (Rev. 4, Ch. 2), Slide 11
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: OUTLINE the unique characteristics of a Pressurizer Vapor Space LOCA.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.2, 41.3, 41.5, and 41.14
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 3 | Tier # | 2 | 2 |
| K/A Statement: Non Nuclear Instruments: Physical connections / cause effect relationship between NNI and the Steam Dump System | Group # | 2 | 2 |
| | K/A # | 016K1.03 | |
| | Importance Rating | 3.2 | 3.2 |

Proposed Question:

With the plant initially at 100% power, and Turbine Impulse Pressure Transmitter 3MSS*PT505 selected at MB7, the following sequence of events occurs:

1. Turbine Impulse Pressure Transmitter 3MSS*PT506 instantly fails to zero.
2. The crew enters AOP 3571, *Instrument Failure Response*.
3. Per AOP 3571, the BOP operator takes the Steam Dump Mode Selector Switch to "RESET."

Which annunciator came in due to the PT506 failure, AND cleared when the BOP operator selected RESET?

a) CONDENSER AVAIL FOR STM DUMP C-9
b) TURBINE BYPASS VV TRIPPED OPEN
c) TURB BYPASS VV ARM FOR OPENING
d) TURB BYPASS T AVE INTLK BYPASSED

Proposed Answer: C

Explanation (Optional):
"C" is correct, since this annunciator is received when the Steam Dump Valves ARM, and PT 506 failing low will generate a C-7 Load Reject signal, arming the dumps (also producing a TURB LOAD REJECTION ARM C-7 annunciator); and selecting RESET removes the arming signal, clearing the annunciators.
"A" is wrong, since 3MSS*PT506 does not input into this annunciator. "A" is plausible, since this alarm would block steam dump operation, and must be cleared in order to restore steam dump operation.
"B" is wrong, since this is driven by PT505, not PT506. "B" is plausible, since this annunciator is received if Tref is significantly below Tave, and PT506 failed low.
"D" is wrong, since this annunciator comes in on Low Tave when the Interlock Selector Switch is selected to Bypass. "D" is plausible, since it is related to Steam Dump switch operation.

Technical Reference(s): AOP 3571 (Rev. 17), Attachment G, step G4 and G.5
(Attach if not previously provided, Functional Sheet 10 (Rev. J)
including version/revision number.) LSK 3-1.1C (Rev. 7) and 3-1.1E (Rev. 8)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure, partial or complete, of the Steam Dump System, determine the effects on the system and on interrelated systems.
Question Source: Bank #405068
Question History: Last NRC Exam Millstone 3 2009 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.7, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 4 | Tier # | 1 | 1 |
| K/A Statement: Reasons for actions contained in | Group # | 1 | 1 |
| the EOP for reactor trip | K/A # | EPE007.EK3.01 | |
| Proposed Question: | Importance Rating | 4.0 | 4.6 |

With the plant at 100% power, the following sequence of events occurs:

1. The reactor trips due to a loss of offsite power.
2. The crew enters ES-0.1, *Reactor Trip Response*.
3. Due to the loss of buses 34A and 34B, ES-0.1 directs the crew to close the MSIVs.

What is the basis for closing the MSIVs during this event?

a) Prevent overheating / overpressurizing the Main Condenser.
b) Prevent spurious secondary plant actuations from distracting the crew while stabilizing the plant.
c) Maintain the secondary plant in a known configuration.
d) Maintain Main Condenser hotwell level stable due to a loss of Main Condenser drawoff capability.

Proposed Answer: A

Explanation (Optional):
On a loss of both non-emergency 4KV buses, all Main Circulating Water Pumps have lost power, resulting in a loss of cooling to the main condenser.
"A" is correct, and "B", "C", and "D" wrong, since the basis for step 3 states that "when both normal busses are deenergized MSIV's are closed…. These actions ensure the condenser is **NOT** overheated / overpressured." During a Millstone 3 LOP event, there was a delay in closing the MSIVs, and the overpressurization resulted in the Main Turbine rupture disks blowing.
"B" and "C" are plausible, since these relate to the basis for closing the MSIVs in EOP 3509.1 during a fire in the control room /IRR/Cable Spreading Area.
"D" is plausible, since on a loss of air, PEOs locally isolate Condenser Drawoff due to loss of Hotwell level indication.

Technical Reference(s): ES-0.1 (Rev. 30), Step 3.n and 3.p
(Attach if not previously provided, ES-0.1 (Rev. 30), BKG doc, Step 3, Justification 2
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: DISCUSS the basis of major procedure steps &/or sequence of steps in EOP 35 ES-0.1.

Question Source: Bank #408749
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4, 41.8, and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 5 | Tier # | 2 | 2 |
| K/A Statement: Emergency Core Cooling System: | Group # | 1 | 1 |
| Predict / monitor avoidance of thermal and pressure stresses due to operating controls to start a pump | K/A # | 006A1.01 | |
| | Importance Rating | 3.1 | 3.4 |

Proposed Question:

A loss of all AC power has occurred, and current plant conditions are as follows:

1. The crew has transitioned to ECA-0.2, *Loss of All AC Power – Recovery with SI Required*.
2. The crew is preparing to start a Charging Pump in the injection mode.
3. Just prior to starting the pump, the crew is directed to check the RCP Seal Supply Isolation Valves (3CHS*MV8809A-D) CLOSED.

What does checking 3CHS*MV8809A-D closed prior to starting the Charging Pump prevent?

a) Hot seal return water from reaching the suction of the Charging Pumps
b) Rad release caused by overflowing the VCT
c) Thermal shock of the RCP seals
d) Steam formation in the Reactor Plant Component Cooling Water System

Proposed Answer: C

Explanation (Optional):
During the loss of all AC power, RCP seal injection from the charging pumps has been lost, so RCP seal supply is from the RCS via the thermal barrier heat exchangers. But these heat exchangers were not being supplied by cool RPCCW flow during the loss of all AC power, so the seal supply water was at RCS temperature, which is very hot.
"C" is correct, since starting a Charging Pump with the seal supply valves open would shock the hot RCP seals with cold RWST water via the Charging Pumps.
"A" is wrong, but plausible, since this is a basis for closing the Seal Return Valve in ECA-0.2.
"B" is wrong, but plausible, since this is a basis for closing the Seal Return Valve in ECA-0.0.
"D" is wrong, but plausible, since this is the basis for closing the RPCCW Containment Isolation Valves in ECA-0.0.

Technical Reference(s): ECA-0.2 (Rev. 14), step 6
(Attach if not previously provided, BKG EOP 35 ECA-0.2 (Rev. 14), step 6 Basis
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 ECA-0.2.

Question Source: Modified Bank #407621 (Parent question included below)
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.3, 41.8, and 41.10

Comments:
This question is considered “modified” since a stem condition has been changed in that it is now after AC Power has been restored, and the crew is in a different procedure, ECA-0.2, preparing to start a Charging Pump, rather than in ECA-0.0, dispatching PEOs to perform local valve manipulations. Also, one distractor has been changed from “Prevent overflowing the RPCCW Expansion Tank” to prevent “Hot seal return water from reaching the suction of the Charging Pumps”.

Original Bank Question:

The plant has experienced a loss of all AC power, and the operators are carrying out the actions of ECA-0.0, *Loss Of All AC Power*.

The US dispatches a PEO to locally close the RPCCW Ctmt Return Outer Isolation Valves (3CCP*MOV 49A and 3CCP*MOV49B).

Why are the operators directed to close these valves?

a) Prevent thermal shock of the RCP seals.

b) Prevent overflowing the RPCCW expansion tank.

c) Protect the RPCCW system from steam formation.

d) Prevent a rad release caused by overflowing the VCT.

Answer: A

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 6 | Tier # | 2 | 2 |
| K/A Statement: Auxiliary Feedwater: | Group # | 1 | 1 |
| Predict impact / mitigate automatic control malfunction | K/A # | 061.A2.05 | |
| Proposed Question: | Importance Rating | 3.1 | 3.4 |

With the plant initially at 100% power, the following sequence of events occurs:

1. The main turbine trips, resulting in a reactor trip.
2. All 3 AFW Pumps are running and supplying AFW flow to the SG's.
3. The crew transitions to ES-0.1, *Reactor Trip Response.*
4. The BOP begins throttling AFW flow.

In regards to OP 3272, *EOP/AOP User's Guide*, complete the following statement regarding the specific restriction on throttling AFW flow, and the basis for the restriction.

The TDAFW and MDAFW flow control valves should be throttled at a rate that is greater than ___(1) seconds over full travel; and the basis of the stroke time requirement is to __(2)__.

| | (1) | (2) |
|---|---|---|
| a) | 15 | avoid water hammer in the feed lines |
| b) | 15 | avoid challenging the TDAFW Pump's discharge relief valve setpoint |
| c) | 10 | avoid water hammer in the feed lines |
| d) | 10 | avoid challenging the TDAFW Pump's discharge relief valve setpoint |

Proposed Answer: B

Explanation (Optional):
This question is considered a K/A match as it tests the Operator's ability to mitigate an automatic control malfunction. The malfunction is the TDAFW Pump's speed control circuit's (Woodward governor) ability to maintain auto speed control. Based on OE, the governor is challenged when Operator's throttle / isolate AFW too quickly. Based on this OE, OP 3272 was revised to limit how fast operators stroke the TDAFW and MDAFW flow control valves. This helps to limit speed oscillations of the TDAFW pump and helps prevent lifting the TDAFW Pump's discharge relief valve (something has occurred at MP3). In accordance with the K/A, the question tests both (1) predict impact (i.e. challenge to the relief valve) and (2) use procedures to mitigate / control consequences (i.e. limit speed of valve travel).
OP 3272 has specific guidance on throttling or isolating AFW Flow (see Attachment 2, section 2.0 "Isolating / Throttling AFW Flow".
"C" and "D" are wrong, since OP 3272 directs that "the TDAFW and MDAFW flow control valves should be throttled at a rate that is greater than 15 seconds over full travel".
"B" is correct, and "A" wrong, since the guidance explains "rapid closure and opening of the throttle valves results in step changes in TDAFW Pump speed and discharge pressure creating the potential for the governor to overcompensate and challenge the discharge relief setpoint." "A" is plausible, since rapid movement of valves is often associated with water hammer events.

Technical Reference(s): OP 3272 (Rev.12) Att. 2 Section 2.0
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: DESCRIBE the operation of the Motor and Turbine Driven Auxiliary Feedwater Control Valves.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7, 41.8, and 41.10

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 7 | Tier # | 2 | 2 |
| K/A Statement: Reactor Coolant Pump: Predict / monitor | Group # | 1 | 1 |
| changes in seal flow and DP based on operating controls | K/A # | 003A1.09 | |
| Proposed Question: | Importance Rating | 2.8 | 2.8 |

A plant cooldown to MODE 5 is in progress per OP 3208, *Plant Cooldown*, and current conditions are as follows:

- The RO has been directed to maintain proper Seal Injection flow to the RCPs during the cooldown and depressurization.
- Seal injection flow has slowly increased toward the top end of its operating band.
- A PEO on a headset has just commenced locally adjusting seal injection flow.
- The RO is monitoring RCP seal injection flow at MB3.

Complete the following two statements.

The upper end of the procedurally allowed band for seal injection flow to a single RCP is___(1)____.
As the PEO adjusts seal injection flow, the RO will observe seal injection flow lowering___(2)____.

| | (1) | (2) |
|---|---|---|
| a) | 10 gpm | for one RCP at a time |
| b) | 13 gpm | for one RCP at a time |
| c) | 10 gpm | for all RCPs at once |
| d) | 13 gpm | for all RCPs at once |

Proposed Answer: B

Explanation (Optional):
As the plant depressurizes, seal injection flow tends to increase, since the Charging Pumps are centrifugal pumps, and a portion of the seal injection flow is routed to the RCS, where pressure is lowering.
"C" and "D" are wrong, since OP 3208 directs the crew to throttle seal injection flow to individual RCPs as necessary during the RCS depressurization to keep the common Charging Header to Seal Injection Valve (3CHS-HCV182) fully open. This is preferred since excessive throttling of 3CHS-HCV182 may lead to failure of the seal injection filter O-rings if the valve is subsequently re-opened.
"C" and "D" are plausible, since the seal injection path can be throttled to all four RCPs at once via the common Charging Header to Seal Injection Valve (3CHS-HCV182), and this is allowed during EOPs and AOPs. Also, as seal injection to one RCP is adjusted, seal injection to the other RCPs will also change, but they will increase as less flow is sent to the affected RCP.
"B" is correct, and "A" wrong, since the RCP seal injection flow target band is between 8 and 13 gpm.
"A" is plausible, since 40 gpm (average of 10 gpm per RCP) is the upper limit for total RCP seal injection based on the Controlled Leakage LCO, but that only applies with the RCS at 2230-2270 psia.

Technical Reference(s): OP 3208 (Rev. 34), steps 3.2.5, 3.2.6, 4.2.4 and 4.4.10
(Attach if not previously provided, OP 3301D (Rev. 23), Section 4.9, including Note 2 prior to 4.9.1
including version/revision number.) P&IDs 103A (Rev. 29), and 104A (Rev. 54)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the purpose and operation of the controls and interlocks associated with the operation of the Reactor Coolant Pumps.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.3, 41.7, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 8 | Tier # | 1 | 1 |
| K/A Statement: Station Blackout: Operate / monitor | Group # | 1 | 1 |
| restoration of power from offsite | K/A # | EPE055EA1.07 | |
| Proposed Question: | Importance Rating | 4.3 | 4.5 |

With the plant initially at 100% power, the following sequence of events occurs:

1. Offsite Power is momentarily lost in the switchyard, resulting in a reactor trip.
2. Both Emergency Diesels fail to start.
3. The crew is preparing to restore power to 34C from offsite power.

How long after the loss of power can the crew RESET the 34C Undervoltage Lockout relay without having to station an operator to hold in the pushbutton on MB8R?

a) 0 seconds
b) 7.5 seconds
c) 4.5 minutes
d) 6 minutes

Proposed Answer: D

Explanation (Optional):
"D" is correct, and "A", "B", and "C wrong, since a 6-minute time delay exists before the UV lockout relay can be reset by pushing the UV Block pushbutton on MB8R.
"A" is plausible, since if Bus 34C had been energized by the EDG (this was expected to occur on the loss of offsite power), this lockout could be reset at any time.
"B" is plausible, since this is how long the "brownout" UV (<90%) must be present to auto-start the EDG with a SIS or CDA present during degraded voltage conditions.
"C" is plausible, since this is how long the "brownout" UV (<90%) must be present to auto-start the EDG with no SIS or CDA present during degraded voltage conditions.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
OP 3343 (Rev. 15), Step 4.8.10, including NOTE
ECA-0.0 (Rev. 40), step 8
GA-3 (Rev. 05), step 3 c and d
LSKs 24-3K (Rev. 11) and 24-3C (Rev. 08)

Proposed references to be provided to applicants during examination: None

Learning Objective: Restore a 4KV Bus to service

Question Source: Bank #402096
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.6 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 9 | Tier # | 2 | 2 |
| K/A Statement: Monitor automatic operation of the | Group # | 2 | 2 |
| Steam Dump System, including steam flow | K/A # | 041A3.03 | |
| Proposed Question: | Importance Rating | 2.7 | 2.8 |

Initial conditions:

- A plant startup in progress per OP 3203, *Plant Startup.*
- Reactor power is 19%.

The following sequence of events occurs:

1. The BOP operator closes the Main Generator output breaker.
2. The BOP operator starts increasing steam flow to the Main Turbine, raising load to 65 MWe.

What is the response of the Condenser Steam Dump valves to this operation?

a) All Steam Dump Valves will remain closed, since they are not armed.
b) All Steam Dump Valves will remain closed, since they were closed just prior to closing the Generator output breaker.
c) The partially open Steam Dump Valves will throttle closed, responding to decreasing main steam header pressure.
d) The partially open Steam Dump Valves will throttle closed, responding to decreasing Reactor Coolant System Tave.

Proposed Answer: C

Explanation (Optional):
"A" is wrong, since Steam dumps are in the Seam Pressure Mode per OP 3203, step 4.3.1, so they are already armed and responding to Main Steam Pressure Transmitter 3MSS-PT507.
"B" is wrong, since the crew has withdrawn control rods in manual to raise RCS Tave, which raises steam pressure, causing Steam Dumps to throttle open, placing "artificial load" on the Steam Dumps.
"C" is correct, and "D" wrong, since in this alignment, the Main Generator can be placed on line with minimal impact on RCS Tave or SG level; since as steam flow increases to the Main Turbine, main steam pressure will decrease, and the Steam Dump Valves automatically throttle closed based on steam pressure decreasing.
"A" is plausible, since power has already been increased to 19%, and at 25% power, Steam Dumps will be placed in the Tave mode.
"B" is plausible, since Steam Dumps will be closed shortly after placing the Generator on line.
"D" is plausible, since this is how Steam Dumps would respond if armed in the Tave mode, and the Tave mode is the normal mode of operation for the Steam Dumps with the plant on line.

Technical Reference(s): OP 3203 (Rev. 29), steps 4.3.1, 4.3.14.f.1)-f.3), 4.3.57.e, and 4.3.64
(Attach if not previously provided, Functional Sheet 10 (Rev. J)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Steam Dump System (when in the Steam Pressure Mode of operation)…

| | | |
|---|---|---|
| Question Source: | Bank #405058 | |
| Question History: | Last NRC Exam | Millstone 3 2002 NRC Exam |
| Question Cognitive Level: | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41.7 and 41.10 | |

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 10 | Tier # | 2 | 2 |
| K/A Statement: Effect of loss or malfunction of the | Group # | 1 | 1 |
| Ctmt Spray System on the Ctmt Cooling System | K/A # | 026K3.01 | |
| Proposed Question: | Importance Rating | 3.9 | 4.1 |

Initial conditions:

- The plant is at 100% power
- Containment Ventilation Systems are in their normal alignments.

A LOCA occurs, and current conditions are as follows:

- SIS and CDA have actuated.
- Containment pressure is 35 psia and increasing.
- Both Quench Spray Pumps have failed to start.
- No operator actions have been taken.

What is the status of Containment Cooling Systems?

a) ALL Safety Related CAR and CRDM Fans have TRIPPED. ALL previously running Non-Safety Related CAR Fans and CRDM Fans remain RUNNING
b) ALL Safety Related Containment Recirculation (CAR) Fans and CRDM Fans remain RUNNING. ALL Non-Safety related CAR and CRDM Fans have TRIPPED.
c) ALL previously running CAR Fans have TRIPPED. ALL previously running CRDM Fans remain RUNNING.
d) ALL previously running Containment Recirculation (CAR) Fans and CRDM Fans remain RUNNING.

Proposed Answer: A

Explanation (Optional): A HI-3 signal is generated at Containment Pressure (2/4) >23 psia which generates a CDA signal. The QSS Pumps should have started on the CDA signal, but have not.
"B" and "D" are wrong, since the CDA signal will trip the safety related CTMT Recirc and CRDM fans whether or not QSS Pumps start, since they have lost cooling.
"A" is correct, since there is no ESF actuation signal which trips the Non-safety related fans.
"C" is wrong, since OP 3313B requires that 2 Containment Recirculation Fans be operating and that both safety related fans should not normally be operated together to prevent overheating the pressurizer cubicle. OP 3313C requires that 2 CRDM fans be operated when required to have them in service but has no restriction on which fans are operated together. "B", "C", and "D" are plausible, since a combination of safety and non-safety fans are initially running, and some will remain running and some will trip.

| | |
|---|---|
| Technical Reference(s): | OP 3313B (Rev. 07-03), Cautions and Notes prior to step 4.1.1 |
| (Attach if not previously provided) | OP 3313C (Rev. 06-03), step 4.1.1 |
| (including version/revision number) | P&ID 153A (Rev. 29) |

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04259 Describe the operation of the following Containment Ventilation System controls and interlocks… Containment Air Recirculation System… Containment Rod Drive Mechanism Cooling System… (As available)

| | |
|---|---|
| Question Source: | Bank #402845 |
| Question History: | Millstone 3 2011 NRC Exam |
| Question Cognitive Level: | Comprehension or Analysis |
| 10 CFR Part 55 Content: | 41.7 |
| Comments: | |

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 11 | Tier # | 2 | 2 |
| K/A Statement: Containment System design feature / | Group # | 1 | 1 |
| interlock which provides for Ctmt Isolation | K/A # | 103K4.06 | |
| Proposed Question: | Importance Rating | 3.1 | 3.7 |

The crew is performing a plant cooldown per OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- Pzr Pressure is 1950 psia.
- The crew has just verified RCS pressure is below P-11, and has completed all associated ESF BLOCKS required by OP 3208.

A steamline break inside Containment occurs, resulting in the following sequence of events:

1. SG A PRESSURE RATE HI annunciator is received on MB5.
2. Pressurizer pressure drops below 1892 psia.
3. CTMT pressure increases above 18 psia.
4. Steam pressure drops below 660 psig.

Assuming operators did not manually actuate SIS, when did an automatic Containment Isolation Phase A (CIA) first actuate during this event?

a) When the SG A PRESSURE RATE HI annunciator lit.
b) When Pzr pressure decreased below 1892 psia.
c) When CTMT pressure increased above 18 psia.
d) When steam pressure decreased below 660 psig.

Proposed Answer: C

Explanation (Optional):
The only automatic actuation of the Containment Isolation Actuation (CIA) signal is directly from an SIS signal. On the plant shutdown, the low RCS pressure reactor trip (1900 psia) is blocked below P-7 (10% power), and when RCS pressure drops below 2000 psia (P-11), the crew manually blocks certain ESF actuation signals.
"A" is wrong, since the low Steamline Pressure SIS (which would generate a CIA signal) was blocked below P-11 (2000 psia), and when this block is inserted, the high steam pressure rate MSI was instated. But the high Steamline Rate signal only provides an MSI signal, not an SIS signal. So no CIA occurs. "A" is plausible, since the low Steamline Pressure SIS (which would generate a CIA signal) was armed until the crew blocked it below P-11.
"B" is wrong, since the RCS low pressure SIS was blocked below P-11. So no CIA occurs. "B" is plausible, since the RCS low pressure SIS was armed until the crew blocked it below P-11.
"C" is correct, since the CTMT Hi 1 pressure SI is still in service, even after the crew initiated blocks when RCS pressure dropped below P-11. So as Ctmt pressure exceeds 18 psia, SIS actuates, and SIS generates a CIA signal.
"D" is wrong, since the crew blocked the Low Steamline Pressure SIS below P-11. "D" is plausible, since the Low Steamline Pressure SIS was armed until the crew blocked it below P-11.

Technical Reference(s): OP 3208 (Rev. 34), step 4.2.5
(Attach if not previously provided, E-0 (Rev. 34), Entry Conditions
including version/revision number.) Functional Sheets 6 (Rev. J) and 8 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following RPS controls and interlocks… CIA Actuation Signals…

Question Source: Bank #406641

Question History: Last NRC Exam Millstone 3 2009 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 12 | Tier # | 1 | 1 |
| K/A Statement: Loss of Emergency Coolant Recirculation: | Group # | 1 | 1 |
| Operate / monitor for desired operating results | K/A # | WE12EA1.3 | |
| Proposed Question: | Importance Rating | 3.1 | 3.4 |

With the plant initially at 100% power, a LOCA occurs, resulting in the following sequence of events:

1. The crew enters ES-1.3, *Transfer to Cold Leg Recirculation* and completes swap over to cold leg recirculation.
2. Shortly afterwards, sump blockage occurs and the crew stops all associated pumps.
3. The crew transitions to ECA-1.1, *Loss of Emergency Coolant Recirculation*.
4. The crew stops both Quench Spray Pumps.
5. The RO notes several CET's are 720 F and rising
6. The crew successfully re-establishes ECCS Flow from the Recirculation Sump using Step 11 of ECA-1.1.

What action is required to be taken by the crew?

a) Transition to E-1, *Loss of Reactor or Secondary Coolant*.
b) Transition to ES-1.3, *Transfer to Cold Leg Recirculation*.
c) Remain in ECA-1.1, *Loss of Emergency Coolant Recirculation*.
d) Go To FR-C.2, *Response to Degraded Core Cooling*

Proposed Answer: C

Explanation (Optional):
ECA-1.1 contains 2 notes prior to Step 1. One note reads: "Containment Sump blockage is NOT recoverable and NO return to the procedure in effect is made. The other note reads: "If Containment Sump blockage is indicated, then CSF Status Trees should be monitored for EAL classification and Functional Response Procedures shall NOT be implemented." Based on application of these two notes (and lack of other transitional steps), "C" is correct, and "A", "B", and "D" wrong.
"A" is plausible, since this action would be taken if sump blockage didn't occur and the crew was successful in ES-1.3.
"B" is plausible, since this was the procedure that was transitioned out of and it provides direction on sump recirculation.
"D" is plausible, since an ORANGE path exists for Core Cooling. Normally, these conditions would require going to FR-C.2.

Technical Reference(s): ECA-1.1 (Rev. 21), Notes prior to step 1
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: DISCUSS conditions which require transition to other procedures from EOP 35 ECA-1.1
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.8 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 13 | Tier # | 1 | 1 |
| K/A Statement: Degraded Core Cooling: Reasons for | Group # | 2 | 2 |
| operating controls to obtain desired operating results | K/A # | WE06EK3.3 | |
| Proposed Question: | Importance Rating | 4.0 | 3.9 |

With the plant initially at 100% power, the following sequence of events occurs:

1. Safety Injection actuates due to a small break LOCA.
2. When the crew is preparing to transition out of E-0, *Reactor Trip or Safety Injection*, the RO reports an orange path condition exists for "Core Cooling".
3. The crew transitions to the appropriate Functional Restoration Procedure (FRP).

Why will this FRP direct the crew to stop one RCP?

a) To minimize heat input to the RCS.
b) To minimize RCS mass loss out the break.
c) To conserve Steam Generator inventory.
d) To preserve one RCP for future use.

Proposed Answer: D

Explanation (Optional):
"D" is correct, and "A", "B", and "C" wrong, since RCPs are running in close to saturated conditions, which is hazardous to the RCPs, and one RCP is stopped to reserve it for potential future use. If conditions degrade and the crew enters FR-C.1, one mitigation strategy will be to start available RCPs to pump crossover leg water into the core.
"A" is plausible, since this is the basis for stopping RCPs in FR-H.1.
"B" is plausible, since this is the basis for stopping RCPs per E-0/E-1 Foldout Page Criteria, and this is only done if a high head injection pump is running. But stopping all 4 RCPs would be stopped. To be in FR-C.2, there is likely a problem with high head injection.
"C" is plausible, since this is the basis for stopping RCPs in FR-H.1.

Technical Reference(s): WOG Background Document, FR-C.2 Step 6 Basis
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 FR-C.2
Question Source: Bank #408134
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 14 | Tier # | 1 | 1 |
| K/A Statement: Large break LOCA: Knowledge of the reasons for the actions contained in the EOP for emergency LOCA (large break) | Group # | 1 | 1 |
| | K/A # | EPE011EK3.12 | |
| | Importance Rating | 4.4 | 4.6 |

Proposed Question:

A large break LOCA occurs, resulting in the following sequence of events:

1. The crew transitions to ES-1.3, *Transfer to Cold Leg Recirculation*.
2. The crew verifies recirculation flow has been established.
3. While completing the Cold Leg Recirculation alignment, the crew separates the gray boot connectors for RWST to CHS pump suction valves 3CHS*LCV112D and 3CHS*LCV112E.

Complete the following statement.

The crew separates the gray boot connector in order to ensure that if a_________.

a) SIS signal is received, 3CHS*LCV112D and E will OPEN/remain OPEN
b) SIS signal is received, 3CHS*LCV112D and E will CLOSE/remain CLOSED
c) VCT Low-Low Level signal is received, 3CHS*LCV112D and E will OPEN/remain OPEN
d) VCT Low-Low Level signal is received, 3CHS*LCV112D and E will CLOSE/remain CLOSED

Proposed Answer: D

Explanation (Optional):
The VCT low-low level signal/interlock functions to automatically open the RWST to CHS Pump Suction Valves 3CHS*LCV112D and 3CHS*LCV112E, and close the VCT to Charging Pump Suction Valves 3CHS*LCV112B and 3CHS*LCV112C on VCT Lo-Lo level to ensure Charging Pump Suction is maintained on loss of VCT level. On a SIS, 3CHS*LCV112D and 3CHS*LCV112E automatically open to align the RWST to the suction of the Charging Pumps as a high head, borated injection source. And as RWST inventory depletes, the crew resets SI and aligns for Cold Leg Recirculation. In this alignment, the Containment Recirculation (RSS) Pumps now supply Charging Pump suction from the Containment Sump, and the suction path from the RWST is isolated to prevent highly radioactive Containment Sump water from backflowing into the RWST, which is outside Containment. In the Cold Leg Recirc alignment, the operators separate the gray boot connectors to remove the VCT level input to 3CHS*LCV112D and E (RWST to CHS pump suction valves), to defeat the VCT Low-Low level auto swap interlock, which would recreate the potential for Ctmt Sump water to enter the RWST.
"D" is correct, and "C" wrong, since disconnecting the gray boot connector ensures the Charging Pump Suction Valves from the RWST can be closed, and will remain closed if a VCT lo-lo level occurs. "C" is plausible, since the VCT lo-lo level signal is being defeated, and on loss of recirc, the crew will align to utilize the remaining RWST water to cool the RCS.
"A" and "B" are wrong, since the gray boot only removes the VCT low-low level signal. "A" and "B" are plausible, since the RWST path automatically aligns on a SIS, but automatic SIS is blocked by P-4 after SIS has been reset.

Technical Reference(s): ES-1.3 (Rev. 20), step 4.b and c
(Attach if not previously provided, ES-1.3, Step Deviation Document (Rev. 020), Basis for step 4
including version/revision number.) P&IDs 104D (Rev. 30), 112A (Rev. 50), 112C (Rev. 38)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basis of major procedure steps and / or sequence of steps in EOP ES-1.3 and ES-1.4.

Question Source: Bank #407932
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7 and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 15 | Tier # | 2 | 2 |
| K/A Statement: Monitor automatic operation of the | Group # | 1 | 1 |
| Component Cooling Water pumps, including interlocks | K/A # | 008A3.02 | |
| Proposed Question: | Importance Rating | 3.2 | 3.2 |

Initial Conditions:

- The plant is at 100% power.
- "A" Charging Pump is in service.
- 3CCE*P1A, Charging Pump Cooling Pump A, is in service.

The following event occurs:

The breaker for 3CCE*P1A trips open.

Complete the following statements related to the status of the component cooling water (CCE) system.

CCE ___(1)____ being provided to the "A" Charging Pump. The position of the CCE suction and discharge cross connect valves can be determined ___(2)____.

| | (1) | (2) |
|---|---|---|
| a) | IS | at MB3 or locally |
| b) | IS | ONLY locally |
| c) | IS NOT | at MB3 or locally |
| d) | IS NOT | ONLY locally |

Proposed Answer: A

Explanation (Optional):
This question is considered a KA match since Millstone 3 has several Component Cooling Water Systems: Reactor Plant Component Cooling Water (RPCCW)-which is being tested on other questions on this exam, Reactor Plant Chilled Water, which also cools primary plant components, Turbine Plant Component Cooling Water (TPCCW) - which is not safety related, and two safety related, system-specific component cooling water systems - the Charging Pump Cooling (CCE) (which this question is testing), and the Safety Injection Pump Cooling (CCI) System. The normal lineup for the CCE Pumps is one pump running and the other pump in standby, with the suction and discharge cross-connect valves open, allowing the running pump to supply both trains. The standby CCE Pump auto-starts on a SIS, LOP, or Low Discharge Header Pressure signal. The suction and discharge cross-connect valves will auto close on a SIS or LOP. Under the postulated conditions, the standby CCE pp will auto start and will supply cooling to the 'A' Charging Pump (C, D are wrong). Additionally, the suction and discharge cross-connect valves are operated / monitored from their control switches on MB3 (A correct, B, D wrong). "B" and "D" are plausible as the cross-connect valves are not prominently located on MB2. "C" and "D" are plausible as the cross-connect valves go closed on a SIS or LOP.

Technical Reference(s): P&ID 105A (Rev. 23)
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the controls and interlocks associated with the following Charging Pump Cooling System … Charging Pump Cooling Pumps… Cross-connect Valves

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 16 | Tier # | 2 | 2 |
| K/A Statement: Incore Temperature Monitoring System: | Group # | 2 | 2 |
| Effect of loss or malfunction of sensors and detectors | K/A # | 017K6.01 | |
| Proposed Question: | Importance Rating | 2.7 | 3.0 |

An inadequate core cooling event is in progress, and current "A" Train ICC Cabinet toggle/thumbwheel positions are as follows:

- CET/MANUAL Toggle Switch: "CET"
- AI/STATUS Toggle: "AI"
- Data Channel Thumbwheel Dial: "00"

The selected CET Data Channel has developed an "open" condition.

What value is flashing on the CET Monitor indication?

a) >5000
b) 1999
c) 1200
d) 0

Proposed Answer: A

Explanation (Optional):
The CET/MANUAL toggle determines the function of the CET temperature display. In the "CET" position, the maximum "in-scan" core exit temperature is displayed.
AI/STATUS Toggle - Determines how the thumbwheel entry will be used. In the "AI" position, the thumbwheel is used to display a single value on the CET LEDs.
The thumbwheel input is activated by the CET/MANUAL switch and the AI/STATUS SWITCH. With toggle switches set to "CET" and "AI", and thumbwheel set at "00", the CETMAX temperature will be displayed on the CET monitor, and the worst available subcooling (from all CET inputs) will be displayed on the SCM monitor display.
"A" is correct, and "B", "C", and "D" wrong, since based on the given toggle/pushbutton positions, CETs rather than RVLMS Thermocouples are being displayed, and a CET data channel open condition is indicated by a flashing value greater than 5000°F.
"B" is plausible, since "1999" displays when an out-of-range condition exists while the subcooling display is selected.
"C" is plausible, since 1200°F is what the actual CET temperatures indicate that require entry into FR-C.1, and this would be true if the failed CETs were automatically removed from scan.
"D" is plausible, since "0" is displayed when CETs are less than 200°F.

Technical Reference(s): OP 3301K (Rev. 09), Section 3
(Attach if not previously provided, OP 3301K (Rev. 09), Section 4.1
including version/revision number.) OP 3301K (Rev. 09), Attachment 5
Proposed references to be provided to applicants during examination: None
Learning Objective: For the following malfunctions, partial or complete, of the ICCM system, determine the effects on the system and on interrelated systems… loss of core exit thermocouple(s)
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.2 and 41.7
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 17 | Tier # | 1 | 1 |
| K/A Statement: Loss of Secondary Heat Sink: | Group # | 1 | 1 |
| Determine / interpret facility conditions and selection of appropriate procedures | K/A # | WE05EA2.1 | |
| | Importance Rating | 3.4 | 4.4 |

Proposed Question:

Initial conditions:

- The crew has entered ECA-2.1, *Uncontrolled Depressurization of All Steam Generators*.
- The TDAFW Pump is the only available Auxiliary Feedwater Pump.
- AFW flow has been reduced to 100 gpm per SG.
- The crew is checking if ECCS flow should be reduced.
- Pressurizer level is empty.
- All four SG Wide Range levels are 20% and decreasing.

The BOP reports the TDAFW pump has just tripped.

What are the first actions required to be taken by the crew?

a) Continue in ECA-2.1, *Uncontrolled Depressurization of All Steam Generators*, and do maintain ECCS pumps running.
b) Continue in ECA-2.1, *Uncontrolled Depressurization of All Steam Generators*, and terminate Safety Injection.
c) Go to FR-H.1, *Response to Loss of Secondary Heat Sink*, and attempt to restore feed flow from the AFW Pumps.
d) Go to FR-H.1, *Response to Loss of Secondary Heat Sink*, and initiate actions to establish bleed and feed of the RCS.

Proposed Answer: D

Explanation (Optional):
"A" and "B" are wrong, since the Caution prior to step 1 of ECA-2.1 informs the operator to go to FR-H.1 if the capability to feed at greater than 530 gpm is lost, so a transition to FR-H.1 is required.
"A" and "B" are plausible, since the operators have previously throttled AFW flow down to 100 gpm/SG, which has brought in a Heat Sink RED path, and the operators were directed not to transition to FR-H.1 in that case. Also, the stem says that crew is checking if ECCS flow should be reduced, and both A" and "B" involve actions related to that step.
"C" wrong, since in FR-H.1 with at least 3 SG wide range levels less than 21% (24% Adv Ctmt), the crew is required to establish bleed and feed cooling of the RCS. "C" is plausible, since transition to FR-H.1 is required, and restoring feed from the AFW Pumps is the first strategy used in FR-H.1 if Feed and Bleed cooling is not currently required.
"D" is correct, since the crew is required to transition to FR-H.1, and with at least 3 SG wide range levels less than 21% (24% Adv Ctmt), the crew is required to establish bleed and feed cooling of the RCS.

Technical Reference(s): ECA-2.1 (Rev. 21), Cautions prior to steps 1 and 2
(Attach if not previously provided, FR-H.1 (Rev. 27), Section 2.2, and Cautions prior to step 1
including version/revision number.) FR-H.1 (Rev. 27), Step 3, and Continuous Action Page step 3
Proposed references to be provided to applicants during examination: None
Learning Objective: Identify plant conditions requiring entry into EOP 35 FR-H.1…
Question Source: Modified Bank #408160 (Parent Question included below)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5 and 41.10

Comments:
This question is considered “modified” since numerous conditions have been removed from the stem, and a Pressurizer level condition has been added.  Also, Distractor “C” has been changed, with the action now being to attempt to restore feed with the AFW pumps.

Original Bank Question:

Initial Conditions:

- The crew has entered ECA-2.1 *Uncontrolled Depressurization of All Steam Generators*.
- The TDAFW Pump is the only available Auxiliary Feedwater Pump.
- AFW flow has been reduced to 100 gpm per SG.
- The crew is checking if ECCS flow should be reduced.
- RCS pressure is 1800 psia and increasing.
- Core exit TCs are at 463°F and stable.
- Containment Temperature is 170°F and stable.
- Wide Range levels in all Steam Generators are 20% and decreasing.

The BOP reports the TDAFW pump has tripped.

What action is required to be taken by the crew?

a) Continue in ECA-2.1 *Uncontrolled Depressurization of All Steam Generators*, and do not stop ECCS pumps.
b) Continue in ECA-2.1 *Uncontrolled Depressurization of All Steam Generators*, and terminate Safety Injection.
c) Go to FR-H.1 *Response to Loss of Secondary Heat Sink*, and attempt to establish flow from either the main feed or condensate pumps.
d) Go to FR-H.1 *Response to Loss of Secondary Heat Sink*, and initiate actions to establish bleed and feed of the RCS.

Answer: D

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 18 | Tier # | 1 | 1 |
| K/A Statement: Inoperable / Stuck Control Rod: | Group # | 2 | 2 |
| Operate / monitor reactor and turbine power | K/A # | APE005AA1.04 | |
| Proposed Question: | Importance Rating | 3.9 | 3.9 |

The plant is at 85% power with a down-power in progress per OP 3204, *At Power Operations*, when the following sequence of events occurs:

1. The RO reports one Control Bank D Rod is misaligned from the bank by greater than 12 steps.
2. The crew enters AOP 3552, *Malfunction of the Rod Drive System*.
3. After one hour, the rod is declared INOPERABLE.
4. Per AOP 3552, Attachment A, "Misaligned Rod", the crew is directed to perform a rapid downpower to reduce power to less than 75% power.
5. The BOP operator commences reducing turbine load.

Complete the following statement.

Per AOP 3552, Attachment A, the RO will__(1)___ to add negative reactivity to the core; and per OP-AP-300, *Reactivity Management*, one "critical parameter" the RO is specifically required to closely monitor is__(2)___.

| | (1) | (2) |
|---|---|---|
| a) | insert Control Rods | AFD |
| b) | insert Control Rods | QPTR |
| c) | borate | AFD |
| d) | borate | QPTR |

Proposed Answer: C

Explanation (Optional):
During the downpower, turbine power is decreasing, so less heat is being removed from the reactor, causing the RCS to heat up. The RO will add negative reactivity to the core to reduce primary power to help maintain the heat balance.
"A" and "B" are wrong, since the RO is required to borate to add negative reactivity to the core.
"A" and "B" are plausible, since inserting control rods would give more immediate response in Tave, and if the misaligned rod was not in control bank D, the crew would have been directed to insert control rods.
"C" is correct, and "D" wrong, since the five critical parameters identified are Reactor power, RCS temperature, program temperature, AFD, and margin to RIL.
"D" is plausible, since QPTR is a concern, especially with a misaligned rod.

Technical Reference(s): AOP 3552 (Rev. 17), Att. A, Step A.4.a-g, especially e and f
(Attach if not previously provided, OP-AP-300 (Rev. 25), Step 5.3.2
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories contained within AOP 3552
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.1, 41.3, 41.5, 41.6, and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 19 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of new and spent fuel movement procedures | Group # | 1 | 1 |
| | K/A # | G2.1.42 | |
| Proposed Question: | Importance Rating | 2.5 | 3.4 |

The plant is in Mode 6 and preparations are being made to do a Core Off-Load. Current conditions are as follows:

- All Refueling personnel are on station.
- The Unit Supervisor is reviewing OP 3210B, *Refueling Operations*, to ensure required communication(s) have been established for CORE ALTERATIONS.

In accordance with OP 3210B, what communication link(s) are required to be in service during CORE ALTERATIONS?

a) The Control Room must be in communication with the person performing a continuous Spent Fuel Pool level watch.
b) The Control Room must be in communication with the personnel at the Containment Refueling Station.
c) The Refueling SRO must be in communication with the Refueling Machine operator.
d) The Refueling SRO must be in communication with the Containment Upender operator.

Proposed Answer: B

Explanation (Optional):
OP 3210B step 4.1.9 requires direct communications between the Control Room and the refueling station prior to and during CORE ALTS ('B' correct. "A", "C", and "D" are wrong). "A" is plausible as this level watch would be required (per 2.1.13) ONLY if SFP level alarms are inoperable. The stem doesn't indicate a problem with SFP level alarms; therefore, this is incorrect. "C" and "D" are plausible as these communication(s) maybe established but are not a requirement for CORE ALTS.

Technical Reference(s): OP 3210B (Rev. 13), step 4.1.9.
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basis of major procedural steps and / or sequence of steps in OP 3210 series procedures.

Question Source: Bank #403364
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10 and 43.7
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 20 | Tier # | 2 | 2 |
| K/A Statement: Fuel Handling Equipment System: | Group # | 2 | 2 |
| Effect of loss of malfunction of Radiation Monitoring Systems on FHS | K/A # | 034K6.02 | |
| | Importance Rating | 2.6 | 3.3 |

Proposed Question:

The crew is preparing to move recently irradiated spent fuel assemblies in the Spent Fuel Pool.

A loss of the___________would require entry into a Tech Spec LCO ACTION STATEMENT during this evolution.

a) Fuel Building Filter Fans
b) Fuel Pool Area Radiation Monitors
c) Spent Fuel Pool Cooling Pumps
d) Spent Fuel Pool Purification Pumps

Proposed Answer: B

Explanation (Optional):
"B" is correct, since the Fuel Pool Area Radiation Monitors are required by Tech Spec LCO 3.3.3.1.
"A" is wrong, since the Fuel Building Filters are not required per Tech Specs. "A" is plausible, since Fuel Building Filters can be aligned to filter the air exhausting from the Fuel Building prior to release to the atmosphere.
"C" is wrong, since the Fuel Pool Cooling Pumps are not required per Tech Specs. "C" is plausible, since the Fuel Pool Cooling Pumps maintain Fuel Pool temperature.
"D" is wrong, since the Fuel Pool Purification Pumps are not required per Tech Specs. "D" is plausible, since the Fuel Pool Purification Pumps maintain Fuel Pool purity.

Technical Reference(s): Tech Spec LCO 3.3.3.1 (Amendment No. 258)
(Attach if not previously provided, Tech Spec Table 3.3-6 (Amendment No. 244)
including version/revision number.) Tech Spec Table 3.3-6 Table Notations (Amendment No. 243)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Radiation Monitoring System, including the basis for each.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7, 41.11, and 41.13 and 43.2
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 21 | Tier # | 2 | 2 |
| K/A Statement: Chemical and Volume Control System: | Group # | 1 | 1 |
| Manually operate / monitor letdown pressure and | K/A # | 004A4.05 | |
| temperature control valves | Importance Rating | 3.6 | 3.1 |

Proposed Question:

The crew is performing a plant cooldown per OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- The Pressurizer is solid.
- Both trains of RHR are in service.

The US directs the RO to perform the following two evolutions at MB3:

1. Manually lower RCS pressure per OP 3208, and after completing this task,
2. Lower letdown temperature using Letdown Temperature Controller 3CHS-TK130.

Complete the following statement.

The RO will lower RCS pressure using the __(1)__ controller, and lower temperature by depressing the __(2)__ arrow on 3CHS-TK130.

| | (1) | (2) |
|---|---|---|
| a) | RHR to Letdown 3CHS*HCV128 | UP |
| b) | RHR to Letdown 3CHS*HCV128 | DOWN |
| c) | Letdown Pressure Control 3CHS*PCV131 | UP |
| d) | Letdown Pressure Control 3CHS*PCV131 | DOWN |

Proposed Answer: D

Explanation (Optional):

During solid plant operation, the RO will decrease letdown flow to increase RCS pressure.
"A" and "B" are wrong, since OP 3208 directs the operators use 3CHS*PCV131 to control pressure. "A" and "B" are plausible, since with RHR in service, 3CHS*HCV128 is in the letdown path upstream of 3CHS*PCV131, and both controllers are on Main Board 3.
"D" is correct and "C" wrong, since the up/down arrows on the controller correspond to temperature rather than RPCCW flow or valve position. So depressing the down arrow is required to lower temperature by increasing RPCCW flow. "C" is plausible, since the up arrow would be correct if the controller directly corresponded to RPCCW flow, or valve position, and there are reverse-acting controllers on the Main Boards (e. g. the Pzr Spray Valve Controllers utilize the down arrow to close the spray valve to raise RCS pressure).

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
- OP 3208 (Rev. 36), step 4.4.14.f.7), and steps 4.4.4.b and c
- OP 3201 (Rev. 36), Precaution 3.5.3
- P&ID 104A (Rev. 54)
- Training LP CHS004C_ILT (Rev. 6, Ch 0), slides 13 and 31

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Controls and Interlocks associated with the following Chemical and Volume Control Systems components… Letdown Pressure Control Valve…

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3, 41.5, 41.7, and 41.10

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 22 | Tier # | 1 | 1 |
| K/A Statement: Loss of Offsite Power: Determine / interpret | Group # | 1 | 1 |
| T-cold and T-hot indicators (Wide Range) | K/A # | APE056AA2.19 | |
| Proposed Question: | Importance Rating | 4.0 | 4.2 |

A reactor trip has occurred due to a loss of offsite power, and current conditions are as follows:

- The crew is performing actions per ES-0.2, *Natural Circulation Cooldown*.
- The crew has been holding RCS cold leg wide range temperature stable for the past hour.
- Core Exit Thermocouples: 402°F
- Thot in all loops: 400°F
- SG pressures: 181 psig

Complete the following statement concerning what RCS WR Tcold will approximately indicate, and what the current Thot trend is if adequate natural circulation cooling is occurring.

WR Tcold indicates___(1)___; and the Thot trend is slowly___(2)___.

| | (1) | (2) |
|---|---|---|
| a) | 374°F | increasing |
| b) | 374°F | decreasing |
| c) | 380°F | increasing |
| d) | 380°F | decreasing |

Proposed Answer: D

Explanation (Optional):
Natural circulation is verified by the following:

- RCS subcooling greater than 32°F
- Steam generator pressures stable or decreasing
- RCS hot leg WR temperatures are stable or decreasing
- Core exit TCs stable or decreasing
- RCS cold leg WR temperatures at saturation temperature for SG pressure

“A” and “B” are wrong, since RCS cold leg WR temperatures will be at saturation temperature for SG pressure, which is 181 psig + 14.7 = 195.7 psia. Per Steam Tables, Tsat for 195.7 psia is 380°F.
“A” and “B” are plausible, since this would be correct if SG pressures were not converted to psia.
“D” is correct, and “C” wrong, since Thot will be slowly decreasing due to lowering decay heat levels.
“C” is plausible, since natural circulation flow is slowly decreasing as decay heat decreases, so T would increase if only flow rate were considered.

Technical Reference(s): ES-0.1 (Rev. 30), Step 9
(Attach if not previously provided, Steam Tables
including version/revision number.)

Proposed references to be provided to applicants during examination: **Steam Tables**

Learning Objective: DESCRIBE the major parameter changes associated with decreased RCS flow rate.

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.10, and 41.14
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 23 | Tier # | 2 | 2 |
| K/A Statement: Instrument Air System: Knowledge of bus power supply to Instrument Air Compressor | Group # | 1 | 1 |
| | K/A # | 078K2.01 | |
| Proposed Question: | Importance Rating | 3.1 | 3.2 |

With the plant initially at 100% power, the following sequence of events occurs:

1. An electrical fault occurs de-energizing 480V load center 32P.
2. An inadvertent CDA occurs.
3. The crew enters AOP 3583, *Inadvertent Containment Depressurization Actuation*.

Which Instrument Air (IAS) and/or Service Air (SAS) Compressor(s) is/are currently energized?

a) The “A” IAS Compressor ONLY
b) The SAS Compressor ONLY
c) The “A” IAS and “B” IAS Compressors
d) The “B” IAS and SAS Compressor breakers

Proposed Answer: B

Explanation (Optional):
“B” is correct, and “A”, “B”, and “D” wrong, since the power supply to the “A” IAS Compressor is 480V Load Center 32P (which was lost) and the “B” IAS Compressor breaker opens on a CDA signal. So the only remaining energized, closed 480V breaker is the SAS compressor supply breaker, which is powered from Load Center 32K, and did not trip open on the CDA signal.
“A” is plausible, since the Load Center that lost power is a “B” train Load Center, and normally, components labeled as “A” are powered from the “A” Train.
“C” is plausible, since the IAS compressors supply safety related equipment, and Service Air does not.
“D” is plausible, since the load center that supplies the “A” IAS compressor is the one that lost power.

Technical Reference(s): OP 3332A-004 (Rev. 06), page 1 of 3
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe operation of plant air systems under the following normal, abnormal, emergency conditions….

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 24 | Tier # | 1 | 1 |
| K/A Statement: Pzr Pressure Control System Malfunction: | Group # | 1 | 1 |
| Reason for isolating Pzr spray following loss of Pzr heaters | K/A # | APE027AK3.01 | |
| Proposed Question: | Importance Rating | 3.5 | 3.8 |

A plant startup is in progress per OP 3203, *Plant Startup*, and Power Range NIS channels indicate as follows:

- PR N41: 8%
- PR N42: 8%
- PR N43: 11%
- PR N44: 8%

The following sequence of events occurs:

1. The PRESSURIZER PRESSURE HI Annunciator is received on MB4.
2. The RO reports the controlling channel of Pzr Pressure has failed high.
3. The RO reports actual RCS pressure indicates 2220 psia and decreasing.
4. The RO reports all Pzr Heaters are OFF, and both Pzr Spray Valves are OPEN.
5. The US chooses to enter AOP 3581, *Immediate Operator Actions*.

What actions will AOP 3581 direct the operators to take, and why?

a) Take the individual Pzr Spray Valve Controllers to "MAN" and CLOSE the Pzr Spray Valves, in order to prevent a Lo Pzr Pressure Reactor trip at 1900 psia.
b) Take the individual Pzr Spray Valve Controllers to "MAN" and CLOSE the Pzr Spray Valves, in order to prevent a Lo Pzr Pressure Safety Injection at 1892 psia.
c) Place the Master Pzr Pressure Controller in "MAN" and adjust output to $\geq$50%, in order to prevent a Lo Pzr Pressure Reactor trip at 1900 psia.
d) Place the Master Pzr Pressure Controller in "MAN" and adjust output to $\geq$50%, in order to prevent a Lo Pzr Pressure Safety Injection at 1892 psia.

Proposed Answer: D

Explanation (Optional):
This question is considered a KA match since the instrument failure causes the Pzr heaters to remain deenergized at a RCS pressure where they are required to be energized, and also causes spray valves to open, requiring the spray path to be isolated. Also, the low initial power level creates a condition-specific reason for requiring the spray path to be isolated.
"A" and "C" are wrong, since the Low Pzr Pressure Reactor Trip is currently blocked below 10% Power on two of four channels (by P-7), and the Lo Pzr Pressure SIS is currently active.
"A" and "C" are plausible, since if power were above 10%, the low Pzr Pressure Reactor Trip would be active, and would be the first automatic RPS/ESF actuation signal to be received on lowering RCS pressure. Also, the Lo Pzr Pressure SIS will be blocked when RCS pressure is reduced below P-11 on a plant cooldown.
"D" is correct, and "B" wrong, since AOP 3581 directs the crew to place the Master Pressure Controller in Manual and adjust to $\geq$50%, in order to stabilize RCS pressure.
"B" is plausible, since AOP 3581 will direct the crew to manually close the spray valves if taking the master pressure controller to manual does not close the spray valves.

Technical Reference(s): AOP 3581 (Rev. 07), Section 2.2, and Attachment E, step E.1 RNO
(Attach if not previously provided, Functional Sheets 6 (Rev. J) and 11 (Rev. H)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: DISCUSS the basis of major precautions, procedure steps, &/or sequence of steps within AOP3581

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 25 | Tier # | 1 | 1 |
| K/A Statement: Loss of Reactor Coolant Makeup: Operate / | Group # | 1 | 1 |
| monitor CVCS charging low flow alarm, sensor, and indicator | K/A # | APE022AA1.02 | |
| Proposed Question: | Importance Rating | 3.0 | 2.9 |

The plant is operating at 100% power with the "B" Charging Pump in service, when the following sequence of events occurs.

1. The CHARG PP FLOW HI/LO annunciator is received on MB3A.
2. Per the ARP, the crew bypasses the flowpath through 3CHS*FCV121 by closing Charging Flow Control Isolation Valve 3CHS*MV8106 and aligning the safety grade charging flowpath.

Complete the following statement.

The CHARG PP FLOW HI/LO annunciator is received at the point when Charging Pump flow decreases below__(1)___; and the crew aligned the safety grade path through__(2)___to bypass 3CHS*FCV121.

| | (1) | (2) |
|---|---|---|
| a) | 25 gpm | 3CHS*HCV190A |
| b) | 25 gpm | 3CHS*HCV190B |
| c) | 10 gpm | 3CHS*HCV190A |
| d) | 10 gpm | 3CHS*HCV190B |

Proposed Answer: A

Explanation (Optional):
"C" and "D" wrong, since the CHARG PP FLOW HI/LO annunciator is received when Charging flow drops below 25 gpm (or greater than 150 gpm).
"C" and "D" are plausible, since 10 gpm is also significantly lower than normal charging flow.
"A" is correct, and "B" wrong, since the ARP directs the operators to align the safety grade path through 3CHS*HCV190A.
"B" is plausible, since the "B" Charging Pump was running, and 3CHS*HCV190B can be aligned in the discharge path of the "B" Charging Pump.

Technical Reference(s): 3353.MB3A (Rev. 05), 4-9 Setpoint
(Attach if not previously provided, 3353.MB3A (Rev. 05), 4-9, step 5.2
including version/revision number.) P&ID 104A (Rev. 54)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the Chemical and Volume Control System under normal, abnormal, and emergency operating conditions
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 26 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of the process for conducting special or infrequent tests | Group # | 2 | 2 |
| | K/A # | G2.2.7 | |
| Proposed Question: | Importance Rating | 2.9 | 3.6 |

The crew is preparing to perform an evolution which is identified as an "Infrequently Conducted or Complex Evolution (ICCE)".

The Shift Manager is reviewing OP-AA-100, *Conduct of Operations*, and OP-AA-106, *Infrequently Conducted or Complex Evolutions*, to ensure ICCE requirements will be complied with while preparing for and performing the evolution.

Which of the following items is **NOT** specifically required in these procedures for ICCE preparation / performance?

a) During preparation, review the need to conduct simulator training for Operations Department and support personnel.
b) During preparation, conduct a brief on management expectations, including criteria for terminating the evolution.
c) During performance, restrict access to the Control Room to minimize distractions.
d) During performance, ensure a hold point is arranged to allow shift turnover.

Proposed Answer: D

Explanation (Optional):
"A" is wrong, since OP-AA-106, step 3.2.2.e.1 requires a review of the need to conduct simulator training for Operations Department and support personnel before the evolution.
"B" is wrong, since OP-AA-106, step 3.2.2.c.8 requires an OMOC management expectations brief for Operations Department and support personnel, which will include criteria for terminating the evolution.
"C" is wrong, since OP-AA-100, Attachment 2, step 3.2 access to the Control Room to be restricted to minimize distractions during an ICCE.
"D" is correct, since OP-AA-100, Attachment 2, step 8.2 directs the crew to delay turnover during an ICCE so as to minimize distractions and enhance continuity. In such cases, the SM or US is responsible for determining the timing of individual watch station relief.
"A", "B", and "C" are plausible, since two other distractors are also required, and "D" is related to a specific requirement about ICCEs, and overtime limits exist to minimize concerns with crew fatigue.

Technical Reference(s): OP-AA-106 (Rev. 11), Steps 1.2, 3.2.2.c.8 and 3.2.2.e
(Attach if not previously provided, OP-AA-100 (Rev. 43), Attachment 2, Steps 3.2 and 8.2
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: As outlined in OP-AA-106, Infrequently Conducted or Complex Evolutions (ICCE), list the responsibilities of the following individuals:
Senior Operations Manager/OMOC, Test Coordinator, Shift Manager

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 27 | Tier # | 1 | 1 |
| K/A Statement: Loss of Source Range Nuclear | Group # | 2 | 2 |
| Instrumentation: Interrelations between loss of SRNI | K/A # | APE032AK2.01 | |
| and power supplies, including proper switch positions | Importance Rating | 2.7 | 3.1 |

Proposed Question:

Initial Conditions:

- Reactor power is in the Source Range at 100 cps.
- The Reactor Trip Breakers are CLOSED.
- All Control Rods are on the bottom.
- The Source Range (SR) NIS Channel N32 Level Trip Switch has been selected to "Bypass" due to I&C testing in progress.

Instrument Power is lost to SR Channel N32 (Control Power is still being supplied to Channel N32).

Do the Reactor Trip Breakers open? Why or why not?

a) Yes, since the trip bypass function is supplied by INSTRUMENT Power.
b) Yes, since power is below P-6, where the trip bypass function IS NOT armed.
c) No, since the trip bypass function is supplied by CONTROL Power.
d) No, since power is below P-6, where the trip bypass function IS armed.

Proposed Answer: C

Explanation (Optional):
Instrument power supplies indication, and also the input to the high level trip bistable when counts exceed the high level trip setpoint of $10^5$ cps. Control power supplies the high level trip bistable, maintaining it in the energized, "not tripped" state; and also the bistable lamps on the SR NIS drawers. Selecting Bypass sends an energized, "not tripped" signal directly to the trip bistable output to RPS from control power.
"C" is correct, and "A" wrong, since while selected to Bypass, Control Power maintains the Trip Bistable output in an energized, "not tripped" condition.
"A" is plausible, since this would be true if Control Power were lost.
"B" and "D" are wrong, since the level trip bypass function is not affected by counts being above or below P-6.
"B" and "D" are plausible, since above P-6, the SR Trip Block pushbuttons on MB4 are armed, allowing the crew to manually block the Source Range trips when power increases above P-6 a reactor startup.

Technical Reference(s): OP 3360 (Rev. 07-09), Caution 2 prior to step 4.3.1, and step 4.3.2
(Attach if not previously provided, OP 3353.MB4C (Rev. 22), 3-1
including version/revision number.) Functional Sheet #3 (Rev. G)

Proposed references to be provided to applicants during examination: None

Learning Objective: For the following conditions, determine the effects on the Nuclear Instrumentation System and on interrelated systems… Source Range instrument failure below P-6…

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.6 and 41.7
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 28 | Tier # | 1 | 1 |
| K/A Statement: Generator Voltage and Electric Grid | Group # | 1 | 1 |
| Disturbances: Knowledge of annunciator alarms, | K/A # | APE077G.2.4.31 | |
| indications, or response procedures | Importance Rating | 4.2 | 4.1 |

Proposed Question:

The plant is initially at 55% power.

The GENERATOR OVER EXCITATION annunciator is received on MB7.

In accordance with the associated ARP, what actions are required to be taken by the crew?

a) Repeatedly cycle the "VOLT REG" switch (MB7) to the LOWER position until the alarm clears. If unsuccessful, trip the reactor and go to E-0, *Reactor Trip or Safety Injection*.
b) Repeatedly cycle the "VOLT REG" switch (MB7) to the LOWER position until the alarm clears. If unsuccessful, trip the main turbine and go to AOP 3550, *Turbine/Generator Trip*.
c) Place and hold the "VOLT REG" switch (MB7) in the LOWER position until the alarm clears. If unsuccessful, trip the reactor and go to E-0, *Reactor Trip or Safety Injection*.
d) Place and hold the "VOLT REG" switch (MB7) in the LOWER position until the alarm clears. If unsuccessful, trip the main turbine and go to AOP 3550, *Turbine/Generator Trip*.

Proposed Answer: A

Explanation (Optional):
"B" and "D" are wrong, since if unsuccessful at restoring main generator voltage with power above P-9 (51% power), the crew is required to trip the reactor and go to E-0.
"B" and "D" are plausible, since tripping the main turbine and going to AOP 3550 would be correct if power were less than P-9.
"A" is correct, and "C" wrong, since in order to reduce excitation voltage, the Voltage Regulator Switch is required to be cycled, rather than held in the "Lower" position, since this switch sends pulse signals when operated. "C" is plausible, since this pulsing signal design is not used for most Raise/Lower switches or pushbuttons at Millstone 3.

Technical Reference(s): OP 3353.MB7C (Rev. 17), 5-5, Corrective Action 1, including Note
(Attach if not previously provided, OP 3353.MB7C (Rev. 17), 5-5, Corrective Action 4
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of main generator, exciter and regulator components controls and interlocks… manual voltage regulator control switch… automatic voltage regulator control switch…

Question Source: Bank #403600
Question History: Last NRC Exam Millstone 3 2009 NRC
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.5, 41.7, and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 29 | Tier # | 2 | 2 |
| K/A Statement: Residual Heat Removal System: Knowledge | Group # | 1 | 1 |
| of how AOPs are used in conjunction with EOPS | K/A # | 005G2.4.8 | |
| Proposed Question: | Importance Rating | 3.8 | 4.5 |

Initial conditions:

- The plant is in MODE 5
- The "A" Train of RHR is in service in the Cooldown Mode.

The following sequence of events occurs:

1. Instrument Air pressure is lost.
2. The crew enters AOP 3562, *Loss of Instrument Air*.
3. The "A" RHR Pump trips.
4. The crew is preparing to enter EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory*.
5. The US desires to continue implementing applicable steps in AOP 3562, since it has guidance on the effect the loss of air is having on the RHR System.

Which procedure implementation strategy complies with OP 3272, *EOP User's Guide* , while completing the desired steps in AOP 3562 as expeditiously as allowed?

a) Complete the applicable steps in AOP 3562, and then enter and complete EOP 3505.
b) Enter and complete all of the steps in EOP 3505, and then complete all of the steps in AOP 3562.
c) Enter EOP 3505, and then perform the actions in EOP 3505 and AOP 3562 in parallel, completing all of the steps in both procedures in the proper step sequence.
d) Enter EOP 3505, and perform the actions in both procedures in parallel, only completing the steps of AOP 3562 necessary to ensure success of EOP 3505.

Proposed Answer: D

Explanation (Optional):
"D" is correct, since it is acceptable to perform the actions of an AOP in parallel with an EOP provided the actions of the EOP receive priority. It is not necessary to perform all steps in the parallel procedure, only those steps necessary to ensure success of the EOP need to be performed.
"A" is wrong, since the steps of the EOP are required to receive priority over the AOP. "A" is plausible, since AOP 3562 was entered first, the US desires to complete the AOP steps as expeditiously as allowed.
"B" is wrong, since performing all of the steps in AOP 3562 in the proper sequence is not required, and will not complete the desired steps of AOP 3562 as expeditiously as allowed. "B" is plausible, since in general, EOP and AOP steps are required to be performed to completion and are required to be performed in the proper sequence.
"C" is wrong, since only those steps of the AOP necessary to ensure success of the EOP need to be performed, and the US desires to complete the AOP steps as expeditiously as allowed.

| | |
|---|---|
| Technical Reference(s): | OP 3272 (Rev. 12), Section 3.8 |
| (Attach if not previously provided, | EOP 3505 (Rev. 17), Section 2.0 |
| including version/revision number.) | AOP 3562 (Rev. 17), Step 11 |

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the usage of abnormal operating procedures while in the emergency operating procedure network.

| | |
|---|---|
| Question Source: | Modified Bank #408104 (Parent Question included below) |
| Question History: | Last NRC Exam N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 55.41.10 |

Comments:
This question is considered “modified” since the EOP has been changed from a Westinghouse ERG-derived procedure to a Millstone-only EOP, and these are differentiated in the EOP Users’ Guide. Also, the need to complete immediate action steps has been removed from the stem. Distractor “B” has been changed to no longer cover the requirement to complete immediate action steps in the EOP prior to completing any actions in the AOP.

Original Bank Question:

With the plant initially at 100% power, the following sequence of events occurs:

1. The crew enters AOP 3562, *Loss of Instrument Air* due to lowering IAS pressure.
2. IAS pressure starts decreasing at a more rapid rate, and the crew preparing to trip the reactor.

The SM is considering how the crew will implement E-0, *Reactor Trip or Safety Injection*, ES-0.1, *Reactor Trip Response*, and AOP 3562 to mitigate the event after the reactor is tripped.

Which procedure implementation strategy is the crew allowed to carry out in this situation?

a) Prior to entering E-0, the US directs the RO by reading out-loud the step of AOP 3562 that operates Pzr heaters, assisting with RCS pressure control. Later, the remaining actions of AOP 3562 are performed in parallel with ES-0.1

b) The US hands AOP 3562 to the extra senior licensed operator, who directs the BOP by reading out loud the steps of AOP 3562 while the US directs the RO to perform the immediate actions of E-0, and then transitions to ES-0.1

c) After exiting E-0, the US directs the RO and the BOP by reading out-loud all of AOP 3562. After the crew completes all of AOP 3562, the crew then enters ES-0.1.

d) After exiting E-0, the US directs the RO and the BOP by reading out-loud steps from ES-0.1 and AOP 3562 in parallel. Only the steps of AOP 3562 that are necessary to ensure success of ES-0.1 are performed, without completing all of AOP 3562.

Correct Answer: D

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 30 | Tier # | 2 | 2 |
| K/A Statement: Physical connections / cause effect | Group # | 1 | 1 |
| relationship between the Service Water System and D/G | K/A # | 076K1.05 | |
| Proposed Question: | Importance Rating | 3.8 | 4.0 |

All AC power is lost, resulting in the following sequence of events:

1. The crew enters ECA-0.0, *Loss of All AC Power*.
2. A PEO is dispatched to locally start the "A" (Orange Train) EDG.
3. The PEO at the diesel reports Diesel Service Water Outlet Valve, 3SWP*AOV39A is closed.

What action, if any, will the PEO take to open the valve per ECA-0.0?

a) The PEO will open a manual air supply valve on the ground level to align air to the valve.
b) The PEO will open a manual air vent valve on the ground level to vent air off of the valve.
c) The PEO will vent air directly off of the 3SWP*AOV39A valve operator on the mezzanine.
d) No PEO action is required, since 3SWP*AOV39A is designed to auto-open when the diesel starts.

Proposed Answer: B

Explanation (Optional):
"B" is correct, and "A", "C", and "D" wrong, since a physical connection is available for the "A" EDG Service Water Supply Valve to allow venting air off of the valve, and this vent valve, which is on the ground level, is required to be opened to vent air off of 3SWP*AOV39A.
"A" is plausible, since several valves require air to open. In ECA-0.0, e.g. PEOs locally align nitrogen to operate the Atmospheric Relief Valves in ECA-0.0.
"C" is plausible, since the "B" EDG does not have a manual vent valve on on the ground level, so Attachment E, step 1 has the operator open its service water supply valve by venting air directly off of the 3SWP*AOV39B valve operator on a mezzanine.
"D" is plausible, since the LOP should have already opened the valve, and this local action is a contingency action required since the valve has not automatically opened.

Technical Reference(s): ECA-0.0 (Rev. 40), Att. E, step E.1 RNO
(Attach if not previously provided, P&ID 133D (Rev. 56)
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the following Service Water System components… EDG Service Water Outlet Valves (SWP*AOV39A/B)…
Question Source: Bank #407614
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4, 41.8 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 31 | Tier # | 3 | 3 |
| K/A Statement: Ability to use radiation monitoring | Group # | 3 | 3 |
| systems, such as fixed radiation monitors and alarms, | K/A # | G2.3.5 | |
| portable survey instruments, equipment, etc. | Importance Rating | 2.9 | 2.9 |

Proposed Question:

With the plant initially at 100% power, the following sequence of events occurs:

1. A spurious reactor trip occurs.
2. Five minutes post trip, the 'A' Steam Generator develops a 200 gpm tube rupture.
3. The US directs the RO to monitor secondary plant radiation monitors to look for indications of a Steam Generator Tube Rupture.

Assuming no operator actions have been taken to realign plant equipment, what is the only radiation monitor available at this point to reliably diagnose the presence of this tube rupture?

a) 3MSS-RE75, "A" Main Steam Line
b) 3ARC-RE21, Condenser Air Ejector
c) 3MSS-RE80A, "A" Steam Generator N-16
d) 3SSR-RE08, Steam Generator Blowdown

Proposed Answer: B

Explanation (Optional):
"B" is correct since ARC21 is available and sensitive enough to detect a post-trip SGTR.
"A" and "C" are wrong, since these monitors detect N-16 gammas, and N-16 is no longer produced post trip. "A" and "C" are plausible, since these detectors will detect a SGTR pre-trip.
"D" is wrong as SSR08 isolates on MDAFW Pump Start Signal, which occurred due to SG Lo-Lo level due to SG shrink on the trip. "D" is plausible, since SSR08 will detect secondary radiation with the plant online, and operators will be directed to un-isolate it at some point after the trip per EOP direction.

Technical Reference(s): Radiation Monitor Manual (Feb 16, 2012), pages 35, 38 41, 58
(Attach if not previously provided, P&ID 123A (Rev 55)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Radiation Monitoring System, including the basis for each.

Question Source: Bank #408457
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.11
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 32 | Tier # | 1 | 1 |
| K/A Statement: Pressurized Thermal Shock: Interrelations between PTS and components and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features. | Group # | 2 | 2 |
| | K/A # | WE08EK2.1 | |
| | Importance Rating | 3.4 | 3.7 |

Proposed Question:

The crew has entered FR-P.1, *Response to Imminent Pressurized Thermal Shock Condition*, and current conditions are as follows:

- RCS temperature is stable.
- RCS pressure is stable.
- The Pressurizer is saturated.
- Only the Control Group of Pressurizer Heaters is energized.
- Narrow range level for intact SGs is 15%.
- AFW flow to each intact SG is 100 GPM.

The crew determines a 1-hour SOAK of the RCS is required.

Per FR-P.1, which of the following evolutions would be allowed in the next hour (Assuming each action would be taken independent of any other action)?

a) Initiate Auxiliary Spray.
b) Initiate a RCS cooldown at a rate not to exceed 50°F per hour.
c) Energize additional Pressurizer Heaters.
d) Increase AFW flow to 300 GPM per SG to raise NR levels to 50%.

Proposed Answer: A

Explanation (Optional):
During the RCS SOAK, operators are prohibited from either raising RCS pressure or lowering RCS temperature for one hour while temperature stress in the vessel wall is allowed to dissipate.
"A" is correct, since this action would lower RCS pressure.
"B" is wrong, since this action would lower RCS temperature. "B" is plausible, since this cooldown rate is below the normal Tech Spec RCS cooldown rate limit, and this will be allowed in FR-P.1 after the SOAK is complete.
"C" is wrong, since this action would raise RCS pressure.
"D" is wrong, since this action would lower RCS temperature.
"C" and "D" are plausible, since each of the four actions affects either RCS temperature or pressure.

Technical Reference(s): FR-P.1 (Rev. 16), Step 23
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories within EOP 35 FR-P.1
Question Source: Bank #408225
Question History: Last NRC Exam Millstone 3 2019 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3, 41.4, 41.7, 41.10, and 41.14
Comments:

| | | | | |
|---|---|---|---|---|
| Examination Outline Cross-reference: | | Level | RO | SRO |
| Question # 33 | | Tier # | 2 | 2 |
| K/A Statement: Pressurizer Level Control System: Manually operate / monitor Charging Pump and flow controls | | Group # | 2 | 2 |
| | | K/A # | 011A4.01 | |
| Proposed Question: | | Importance Rating | 3.5 | 3.2 |

Initial conditions:

- The plant is in MODE 3
- Pressurizer level is 25%.

The following sequence of events occurs:

1. The US directs the RO to raise Pzr level to 30% using either the Charging Line Flow Controller (3CHS-FK121) or the Pzr Master Level Controller (3CHS-LK459) on MB3.
2. The RO reports the Charging Line Flow Controller (3CHS-FK121) will NOT shift to MANUAL.

Complete the following statement.

The RO will place 3CHS-LK459 in ___(1)___, and monitor Charging flow (gpm) response on___(2)___.

| | (1) | (2) |
|---|---|---|
| a) | "AUTO" and adjust the Level setpoint to 30% | 3CHS-LK459 |
| b) | "AUTO" and adjust the Level setpoint to 30% | 3CHS-FK121 |
| c) | "MAN" and depress the Up Arrow | 3CHS-LK459 |
| d) | "MAN" and depress the Up Arrow | 3CHS-FK121 |

Proposed Answer: D

Explanation (Optional):
The Pzr Master Level Controller (3CHS-LK459) output is based on controlling Pzr level, while the Charging Line Flow Controller (3CHS-FK121) output is designed to control Charging Flow.
"A" and "B" are wrong, since the program Pzr Level setpoint is automatically adjusted based on RCS Tave, and the program setpoint cannot be adjusted on this controller. Pressing the Up Arrow is required for LK459 to increase its output to send a signal to raise Pzr level, which will throttle the Charging Line Flow Control Valve 3CHS*FCV121 open.
"A" and "B" are plausible, since most OIM controllers have a knob that allows adjusting the automatic setpoint.
"D" is correct, and "C" wrong, since Charging flow is indicated on the 3CHS-FK121 controller, but 3CHS-LK459 displays Pzr level. "C" is plausible, since both controllers adjust Charging flow.

Technical Reference(s): P&IDs 104A (Rev. 54) and 102C (Rev. 27)
(Attach if not previously provided, AOP 3571 (Rev. 17), Att. C, steps C.1-C.3
including version/revision number.) Training LP CHS004C_ILT (Rev. 6, Ch 0), slides 13 and 74
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the Controls and Interlocks associated with the following Chemical and Volume Control Systems components… Charging Line Flow Control Valve…
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 34 | Tier # | 2 | 2 |
| K/A Statement: Chemical and Volume Control System: | Group # | 1 | 1 |
| Operational implications of the concept of Shutdown Margin | K/A # | 004K5.19 | |
| Proposed Question: | Importance Rating | 3.5 | 3.9 |

With the plant initially at 50% power, a transient occurs, resulting in the following sequence of events:

1. Letdown temperature entering the CVCS demineralizers increases by 10°F.
2. Several minutes later, the reactor trips.

What effect, if any, did the higher temperature water entering the CVCS demineralizers have on Shutdown Margin now that the plant is in MODE 3?

a) No effect on Shutdown Margin assuming the crew maintained RCS Tave stable prior to the trip.
b) No effect on Shutdown Margin assuming the crew did not move Control Rods prior to the trip.
c) Shutdown Margin INCREASED.
d) Shutdown Margin DECREASED.

Proposed Answer: C

Explanation (Optional):
The definition of Shutdown Margin is the amount by which the reactor is or would be shutdown from its present condition, assuming all rods are fully inserted, except for the most reactive rod, which is assumed to be fully withdrawn. Increasing temperature of the water entering the mixed beds causes boron to be released by the beds, raising boron concentration in the RCS. This adds negative reactivity to the core, which will be offset either by positive reactivity from power defect as power decreases, or positive reactivity from control rod withdrawal if Operators maintain Tave stable.
"A" and "B" are wrong, since RCS boron concentration has increased, and whether temperature or rod position changed prior to the trip, the combined amount of reactivity inserted from rods and power defect on the trip is going to be the same.
"A" and "B" are plausible, since moving rods or allowing Tave to change affects reactivity in the core. Also, in MODE 1, Shutdown Margin is surveilled by verifying rods are above RIL. But this assumes rods would insert in response to a dilution of the RCS.
"C" is correct and "D" wrong, since RCS boron concentration has increased. This raises the amount by which the reactor will be shutdown on the trip, so by definition, SDM has increased. "D" is plausible, since the effect of heating the demineralizer on boron concentration is not described in the stem of the question, and the response of Tave/rod position to the boration is to add positive reactivity to the core.

Technical Reference(s): Tech Spec Definition 1.30 (Amend. 242), SHUTDOWN MARGIN
(Attach if not previously provided, RE Curve and Data Book, Curve RE-B-02 (MP3-21-00)
including version/revision number.) GFS

Proposed references to be provided to applicants during examination: None

Learning Objective: … determine how changes in the variable reactivity parameters will affect shutdown margin.

Question Source: Modified Bank#406704 (Parent question included below)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.1 and 41.3

Comments:
This question is considered “modified” since a reactor trip has been added to the stem.  Also, distractor “A” has been changed to consider Tave change instead of rod position.

Original Bank Question:

The plant is at 50% power.

Due to a malfunction, the temperature of the letdown flow entering the mixed bed demineralizers increases by 10$^{\circ}$F.

What effect, if any, does higher temperature water entering the mixed beds have on Shutdown Margin?

a) No effect on shutdown margin if rods are in Manual.

b) No effect on shutdown margin if rods are in Auto.

c) Shutdown margin INCREASES with rods in Auto or Manual control.

d) Shutdown margin DECREASES with rods in Auto or Manual control.

Correct answer: C

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 35 | Tier # | 2 | 2 |
| K/A Statement: Process Radiation Monitoring System: | Group # | 1 | 1 |
| Knowledge of PRM System design features /interlocks which | K/A # | 073K4.01 | |
| provide for release termination when radiation exceeds setpoint | Importance Rating | 4.0 | 4.3 |

Proposed Question:

The plant is initially at 100% power with the Blowdown System aligned for open cycle blowdown.

A high radiation alarm is received on Steam Generator Blowdown Radiation Monitor, 3SSR-RE08.

How does the SG Blowdown (BDG) System respond to the 3SSR-RE08 alarm?

a) All SG Blowdown Flow Control Valves (3BDG-HV20A-D) CLOSE.
b) All SG Blowdown CTMT Isolation Valves (3BDG*CTV22A-D) CLOSE.
c) The SG Blowdown Tank (3BDG-TK1) vent path realigns to the Main Condenser.
d) The SG Blowdown Tank (3BDG-TK1) drain path realigns to the Main Condenser.

Proposed Answer: B

Explanation (Optional):
"B" is correct, since 3SSR-RE08 isolates the S/G blowdown CTMT Isolation Valves on a high radiation signal.
"A", "C", and "D" are wrong, since these valves do not close on a high radiation signal.
"A", "C", and "D" are plausible, since realigning these valves would terminate a release.

Technical Reference(s): AOP 3573 (Rev. 28), Attachment A, Page 13 of 13
(Attach if not previously provided, AOP 3573 (Rev. 28), Definition 2.5.2
including version/revision number.) P&ID 123A (Rev. 61)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following Radiation Monitors controls and interlocks… SSR-RE08

Question Source: Bank #404669
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.11 and 41.13
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 36 | Tier # | 1 | 1 |
| K/A Statement: Loss of Containment Integrity: Knowledge of the parameters and logic used to assess the status of safety functions, such as… containment conditions… | Group # | 2 | 2 |
| | K/A # | APE069G.2.4.21 | |
| | Importance Rating | 4.0 | 4.6 |

Proposed Question:

The crew is performing actions in E-1, *Loss of Reactor or Secondary Coolant*.

The RO reports the following parameters associated with the Containment CSF Status Tree:

- Containment pressure is 24 psia and slowly increasing.
- Containment radiation is 20 R/hr and slowly increasing.
- Containment sump level is 15 feet and slowly increasing.
- No Quench Spray Pumps are running.

Which Orange path, if any, exists for the Containment CSF Status Tree?

a) High Containment Pressure
b) Containment Flooding
c) High Containment Radiation
d) None. The Status Tree is Yellow

Proposed Answer: A

Explanation (Optional):
"A" is correct and "D" wrong, since one CTMT orange path is from CTMT pressure of 23 psia with no CTMT Spray Pumps running (This condition currently exists).
"D" is plausible, since if a CTMT Spray Pump were running, this would be the correct answer.
"B" is wrong, since the CTMT flooding orange path threshold is high sump level of 15.75 feet, which is higher than the current sump level of 15 feet. "B" is plausible, since CTMT Sump level is elevated.
"C" is wrong, since the highest priority status tree color for CTMT radiation is yellow. "C" is plausible, since current radiation (20 R/hr) is above the yellow path setpoint of 10R/hr.

Technical Reference(s): EOP 35 F-0.5 (Rev. 04), Containment CSF Status Tree
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a set of plant conditions and CSF status trees, DETERMINE the status of the CSFs
Question Source: Bank #408312
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10 and 43.5
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 37 | Tier # | 1 | 1 |
| K/A Statement: Determine / interpret failure modes of | Group # | 1 | 1 |
| air-operated equipment on a loss of Instrument Air | K/A # | APE065AA2.08 | |
| Proposed Question: | Importance Rating | 2.9 | 3.3 |

The plant is in MODE 5, at the end of a refueling outage, and initial conditions are as follows:

- The Pressurizer is solid.
- The "A" RHR train is in service.
- RCS Temperature is stable at 145$^{o}$F.
- RHS*FCV 606 is 10% open.

The following sequence of events occurs:

1. A large crack is discovered on the Instrument Air (IAS) header in the ESF Building.
2. Operators isolate instrument air to the ESF Building.
3. IAS pressure depressurizes to zero psig in the ESF Building.

Assuming no other operator actions are taken, complete the following statement about RCS temperature response:

RCS temperature will___(1)____due to___(2)____flow through the RHR Heat Exchanger.

| | (1) | (2) |
|---|---|---|
| a) | increase | decreased RPCCW |
| b) | increase | decreased RHR |
| c) | decrease | increased RPCCW |
| d) | decrease | increased RHR |

Proposed Answer: D

Explanation (Optional):
Post-refueling decay heat is minimal, so 3RHS*HCV606 is initially throttled mostly closed.
"A" and "C" are wrong, since a loss of IAS will cause 3CCP*FV66A to fail AS IS, resulting in NO change in CCP flow.
"D" is correct, and "B" wrong, since RHR Flow Control Valve 3RHS*HCV 606 fails open on a loss of IAS, resulting in maximum RHR flow through the RHR HX.
"A", "B", and "C" are plausible, since there are air-operated valves in both the RHR System and the RPCCW System. Also, for each distractor, as flow changes, RCS temperature responds in the appropriate direction based on the assumed fail position of the associated valve.

Technical Reference(s): AOP 3562 (Rev. 17), steps 2.f, and 20.a
(Attach if not previously provided, AOP 3562 (Rev. 17), Att. C, pages 3 and 4 of 5
including version/revision number.) P&IDs 112A (Rev. 50) and 121A (Rev. 34)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure, partial or complete, of plant air systems, determine effects on the systems and interrelated systems
Question Source: Bank #404612
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7 and 41.8
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 38 | Tier # | 1 | 1 |
| K/A Statement: Loss of Vital AC Electrical Instrument Bus: | Group # | 1 | 1 |
| Reasons for actions contained in the EOP for loss of vital ac electrical instrument bus | K/A # | APE057AK3.01 | |
| | Importance Rating | 4.1 | 4.4 |

Proposed Question:

With the plant at 100% power, the following sequence of events occurs:

VIAC 1 deenergizes.
Letdown automatically isolates.
The crew enters AOP 3564, *Loss of One Protective System Channel*.

Per AOP 3564, *Loss of One Protective System Channel,* Step 2 RNO, the crew ensures Charging is isolated.

Why is this step designated as a Time Critical Operator Action?

a) Prevent a loss of Shutdown Margin due to the potential for an inadvertent dilution.
b) Prevent a RCS boration due to the Charging Pump suctions swapping over to the RWST.
c) Prevent water relief through the Pzr PORVs.
d) Prevent water relief through the Pzr Safety Valves.

Proposed Answer: D

Explanation (Optional):
"D" is correct, and "A", "B", and "C" wrong, since on a loss of VIAC 1, Pzr level fails low, resulting in letdown isolation and Charging flow increasing to maximum. This is the worst-case RCS mass addition event, and operators have 8.4 minutes to either terminate the event (step 2 accomplishes this action), or verify the PORV Block Valves are open, ensuring the PORVs are available to prevent water relief through the Pzr Safety Valves, since the Pzr Safety Valves are not qualified to pass water.
"A" is plausible, since instruments have failed, and failure of the Makeup System in the dilution mode is a worst-case dilution event.
"B" is plausible, since one of the instruments that may lose power on a loss of VIAC is VCT level transmitter 3CHS-LT112, which controls VCT makeup. But this is covered in a separate step in AOP 3564, which has the crew align for manual VCT makeup, if required.
"C" is plausible, since Pzr overfill is the concern, but the PORVs are qualified to pass water for one hour.

Technical Reference(s): AOP 3564 (Rev. 13), step 2, and Attachment A
(Attach if not previously provided, BKG AOP 3564 (Rev. 13), Step 2 and Attachment A
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basis of major precautions, procedure steps and/or sequence of steps within AOP 3564, Loss of One Protective System Channel.

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.3, 41.7, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 39 | Tier # | 1 | 1 |
| K/A Statement: Steam Generator Tube Leak: | Group # | 2 | 2 |
| Operational implications of leak rate vs. pressure drop | K/A # | APE037AK1.02 | |
| Proposed Question: | Importance Rating | 3.5 | 3.9 |

With the plant initially at 10% power, a tube leak initiates in the "B" SG, resulting in the following sequence of events:

1. The crew enters AOP 3576, *Steam Generator Tube Leak*.
2. The crew shuts down the plant due to the size of the leak.
3. The crew stops feeding the "B" SG.
4. The crew depressurizes the RCS to match "B" SG pressure.

Current Conditions:

- The crew is controlling RCS pressure and charging flow to minimize RCS-to-secondary leakage.
- PZR level is 48% and stable.
- "B" SG NR level is 64% and slowly increasing.

What action is required to be taken by the crew to terminate the rising level in the "B" SG?

a) Increase RCS pressure using Pzr Heaters
b) Decrease RCS pressure using Pzr Spray
c) Increase Charging flow
d) Decrease Charging flow

Proposed Answer: B

Explanation (Optional):
PZR level is in its desired band, but RCS pressure has increased above ruptured SG pressure, as evidenced by the increasing SG level. Since the crew is not feeding the "B" SG, the increasing level is due to RCS leakage into the SG.
"B" is correct, and "A", "C", and "D" wrong, since the crew will lower RCS pressure to stop the leakage. "A", "C", and "D" are plausible, since these are all procedurally directed actions based on the status of the PZR and the ruptured SG.

Technical Reference(s): AOP 3576 (Rev. 09), Step 25
(Attach if not previously provided, BKG AOP 3576 (Rev. 09), Steps 19, 20, and 25
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories within AOP 3576
Question Source: Bank #407286
Question History: Last NRC Exam Millstone 3 2007 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3, 41.4, 41.5 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 40 | Tier # | 2 | 2 |
| K/A Statement: AC Distribution System: Design feature / interlock for one-line diagram of 4kV to 480V distribution, including sources of normal and alternative power | Group # | 1 | 1 |
| | K/A # | 062K4.07 | |
| | Importance Rating | 2.7 | 3.1 |

Proposed Question:

The plant is in MODE 5.

Maintenance is required on the supply transformer to Load Center 32R.

How, if at all, can power be aligned to 32R prior to deenergizing the transformer?

a) 32R can be cross tied to 480V bus 32D
b) 32R can be cross tied to 480V bus 32W
c) 32R can be cross tied from 480V bus 32S or 32Y
d) 32R cannot be cross-tied to another 480V bus

Proposed Answer: C

Explanation (Optional):
"C" is correct, and "D" wrong, since 32R can be cross tied to emergency buses from the same train (32S or 32Y). "D" is plausible, since 120 Volt AC Buses and 125 V DC Buses cannot be cross tied to another bus.
"A" is wrong, since emergency load centers cannot be cross tied to non-emergency load centers. "A" is plausible, since 32D is from the same train.
"B" is wrong, since emergency load centers cannot be cross tied to the opposite train. "B" is plausible, since 32W is an emergency bus, and cross-tying to the opposite train is the cross-tie method for non-emergency load centers.

Technical Reference(s): EE-1A (Rev. 27)
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: Describe the 480 VAC distribution system operation under… Cross-tie operations
Question Source: Bank #402027
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge.
10 CFR Part 55 Content: 55.41.7 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 41 | Tier # | 2 | 2 |
| K/A Statement: Main and Reheat Steam System: Ability to verify that alarms are consistent with plant conditions | Group # | 1 | 1 |
| | K/A # | 039G2.4.46 | |
| Proposed Question: | Importance Rating | 4.2 | 4.2 |

The plant was at 100% power when a reactor trip occurs due to a turbine trip.

Ten minutes after the trip, the crew is reviewing all lit annunciators, and the BOP reports the following annunciators are lit:

- MB5A 5-5: TDFW PP A TURBINE TRIP
- MB5C 5-3: TDFW PP B TURBINE TRIP
- MB6A 1-1B: DWST LEVEL LO
- MB6A, 3-4: FEEDWATER HEATER LEVEL LO
- MB6A, 5-8: 4TH POINT HEATER LEVEL LO
- MB6B 2-8A: MOIST SEP DRN TK A LEVEL HI
- MB7B 5-1: MOIST SEP WATER LEVEL HI

Which of these annunciators is NOT expected at this point in ES-0.1, *Reactor Trip Response*?

a) The TDFP TRIP annunciators should not be lit, since no signal trips them if SIS does not occur. The crew needs to check the Heat Sink status tree.
b) The DWST LEVEL LO annunciator should not be lit, since the AFW recirculation path is always aligned to the DWST. Aux Feed Pump suctions must be swapped to the CST.
c) The FEED HEATER LEVEL LO annunciators should not be lit, since the main feed pump recirc valves should have opened on the Feedwater Isolation. Operators need to trip the running 4th Point Heater Drain Pumps.
d) The MOIST SEP LEVEL HI annunciators should not be lit, since the "A" Moisture Separator Drain Tank Emergency Drain Valve should have opened. This may have been the cause of the turbine trip.

Proposed Answer: D

Explanation (Optional): "D" is correct, since high level in the MSR Drain Tank sends a signal to open the emergency drain valve. If level continues to increase, the MOIST SEP WATER LEVEL HI results in a turbine trip. This protects the turbine from water entering from the Moisture Separator.
"A" is wrong, since the TDMFPs receive a trip signal after a time delay indirectly from P-4, preventing feed header overpressurization on a trip. "A" is plausible since the trip is indirect, after a time delay.
"B" is wrong, since AFW takes a suction on the DWST, lowering DWST level. "B" is plausible, since AFW swapover to the CST is required on DWST LO-LO level, and AFW pumps recirc to the DWST.
"C" is wrong, since the source of extraction steam to the heaters (the HP and LP turbines) have been isolated on the turbine trip. So heater levels drop and low level alarms are expected. "C" is plausible, since the heater strings remain unisolated on a reactor trip.

Technical Reference(s): P&ID 125A (Rev. 35)
(Attach if not previously provided, including version/revision number.)
OP3353.MB5A 5-5 (Rev. 12), MB5C, 5-3 (Rev. 12)
OP3353.MB6A 1-1B (Rev. 17), MB6A 3-4 (Rev. 17)
OP 3353.MB6A 5-8 (Rev. 17), MB6B 2-8A (Rev. 00)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure, partial or complete, of the moisture separator reheater system, determine the effects on the system and on interrelated systems

Question Source: Bank #408728
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.5, 41.7, and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 42 | Tier # | 2 | 2 |
| K/A Statement: Condenser Air Removal System: | Group # | 2 | 2 |
| Ability to interpret and execute procedure steps | K/A # | 055G2.1.20 | |
| Proposed Question: | Importance Rating | 4.6 | 4.6 |

Over the course of a month, the crew performs the following four sections from OP 3329, *Condenser Air Removal*.

Section 4.1, "Place Mechanical Vacuum Pumps in Service"
Section 4.5, "Place a Second Steam Jet Air Ejector in Service"
Section 4.7, "Align Steam Jet Air Ejector Exhaust Flow to Turbine Building"
Section 4.9, "Fill Vacuum Breakers 3ARC-MOV20A, 3ARC-MOV20B, and 3ARC-MOV20C"

Which of these sections contains steps that require valve manipulation from the Main Boards?

a) Section 4.1
b) Section 4.5
c) Section 4.7
d) Section 4.9

Proposed Answer: A

Explanation (Optional):
"A" is correct, since placing Mechanical Vacuum Pumps in service requires the crew to check the Condenser Vacuum Breakers (3ARC-MOV20A/B/C) closed at MB7, open the Condenser Vacuum Suction Valves (3ARC-AOV27A/B) at MB7, and start the Condenser Vacuum Pumps (3ARC-P1A/B).
"B", "C", and "D" are wrong, since all of the steps in these procedure sections are performed locally in the Turbine Building.
"B", "C", and "D" are plausible, since all four of these procedure sections contain several steps that are performed locally, and this system is rarely operated, except when starting up or shutting down the plant.

Technical Reference(s): OP 3329 (Rev. 12), Sections 4.1, 4.5, 4.7, and 4.9
(Attach if not previously provided, P&ID 127B (Rev. 16)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Condenser Air Removal system under normal and abnormal conditions

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 43 | Tier # | 2 | 2 |
| K/A Statement: DC Electrical Distribution: | Group # | 1 | 1 |
| Design feature / interlock which provides for trips | K/A # | 063K4.04 | |
| Proposed Question: | Importance Rating | 2.6 | 2.9 |

With the plant initially operating at 100% power, the following sequence of events occurs:

1. Offsite power is lost.
2. Thirty-five (35) minutes later, power has NOT been restored to Non-Emergency 4160 Volt Buses 34A and 34B.

Assuming no operator actions have been taken with the electric plant, what is the status of Instrument Bus 6 (3VBA-PNL6A/B)?

a) Energized from Inverter 6, which is being supplied by its MCC 32-3T.
b) Energized from Inverter 6, which is being supplied by Battery 6.
c) Deenergized, due to the Battery Bus to Inverter 6 DC Input Breaker being tripped open.
d) Deenergized, due to the Inverter 6 AC Output Breaker to Instrument Bus 6 being tripped open.

Proposed Answer: C

Explanation (Optional):
Both the normal and alternate AC power sources to 3VBA-PNL6A/B are supplied from 480 VAC MCC 32-3T. Upon a loss of offsite AC power, the supply breaker from safety related 480V Load Center 32T to non-safety related 480V MCC 32-3T trips open to protect the emergency bus. At that point, Battery Bus 6 will supply Inverter 6, which will continue to supply 3VBA-PNL6A/B. In order to extend battery life, once current from the battery bus is sensed, a 30-minute timer will begin to time out. After 30 minutes of battery operation, the DC battery supply breaker to Inverter 6 will trip open.
"C" is correct and "A" and "B" wrong, since more than 30 minutes have passed, and when the battery bus supply breaker to the inverter triped open with 32-3T deenergized, 3VBA-PNL6A/B lost power.
"A" is plausible, since this is the correct lineup 35 minutes after a LOP for the safety related VIACs, and MCC 32-3T is normally powered from Emergency Load Center 32T.
"B" is plausible, since this is the lineup to 3VBA-PNL6A/B for the first 30 minutes following a LOP.
"D" is wrong, since the Battery Bus 6 to Inverter 6 Supply Breaker trips open after 30 minutes, not the Inverter 6 AC Output Breaker. "D" is plausible, since opening the Inverter 6 AC Output Breaker would also deenergize 3VBA-PNL6A/B during a LOP, since 32-3T supplies its alternate source, and 32-3T is denergized.

Technical Reference(s): OP3345A (Rev. 013), Precaution 3.1
(Attach if not previously provided, E-0 Basis Document (Rev. 34-00), Step 27 basis, page 58
including version/revision number.) EE-1U (Rev. 19) and EE-1BA (Rev. 31)
Proposed references to be provided to applicants during examination: None
Learning Objective: DESCRIBE the 125 VDC distribution system operation under normal, abnormal, and emergency conditions… Loss of off-site power
Question Source: Bank #401905
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 44 | Tier # | 2 | 2 |
| K/A Statement: Control Rod Drive System: Physical connections / cause effect relationship between CRDS and CRDM | Group # | 2 | 2 |
| | K/A # | 001K1.03 | |
| Proposed Question: | Importance Rating | 3.4 | 3.6 |

Complete the following statement about how the Control Rod Disconnect Switches interface between the Rod Control System and the CRDMs.

A specific Control Rod Disconnect Switch connects/disconnects power____________the associated rod.

a) to the Movable Gripper Coil for
b) to the Lift Coil for
c) from the Hold Bus to
d) from the Logic Cabinet output to

Proposed Answer: B

Explanation (Optional):
"B" is correct, since the Rod Disconnect Switches are used to disconnect power to the Lift Coils for the operable rods while recovering a dropped or misaligned rod. This ensures the remaining rods do not move while restoring the affected rod. Disconnecting power to the Lift Coils allows for normal step sequencing without actually moving the rod, as the associated rod simply transfers between the stationary and movable grippers, without actually lifting or lowering.
"A" is wrong, since deenergizing a movable gripper would cause rod to drop when stationary gripper released. "A" is plausible, since normally, this Coil is deenergized, and a rod cannot be moved if this gripper does not engage.
"C" is wrong, since entire banks are placed on the hold bus, not just one rod. "B" is plausible, since placing a rod on the hold bus will not allow the rod to be moved.
"D" is wrong, since the logic cabinet sends output to power cabinets, not individual rods. "D" is plausible, since the logic cabinet determines the required step sequence signal to be sent to the rods, and if this were interrupted, the rod would remain stationary.

Technical Reference(s): AOP 3552 Bkg Doc (Rev. 17-00), Att. A, top of page 20 of 35
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the function and location of the following Rod Control System components… Control Rod Disconnect Switches
Question Source: Bank #404768
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.6
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 45 | Tier # | 2 | 2 |
| K/A Statement: Physical connections / cause effect | Group # | 1 | 1 |
| relationship between Ctmt and the Ctmt Cooling System | K/A # | 103K1.01 | |
| Proposed Question: | Importance Rating | 3.6 | 3.9 |

With the plant at 100% power, one of the two running Reactor Plant Chilled Water (CDS) pumps trips, resulting in one of the running chillers to trip as well.

Assuming no operator actions are taken, what will be the first limit reached during this event?

a) CRDM high temperature limit
b) Reactor Coolant Pump high temperature limit
c) Ctmt high pressure Tech Spec limit
d) Neutron Shield Tank high temperature Tech Spec limit

Proposed Answer: C

Explanation (Optional):
As CDS heats up, its cooling water supplied to the Containment Cooling Systems heats up. This affects the Containment Air Recirculation (CAR) System, the CRDM Cooling System, and the Neutron Shield Tank. With decreased cooling to CTMT, CTMT temperature increases, which raises CTMT pressure.
"C" is correct, since CTMT pressure has a limited margin before Tech Specs require action.
"D" is wrong, since Neutron Shield Tank temperature is not covered by Tech Specs. "D" is plausible, since the Neutron Shield Tank is cooled by Reactor Plant Chilled Water.
"A" and "B" are wrong, since RCPs and CRDMs will not experience immediate problems, since the associated coolers cool the hot air exhausting from these heat loads to assist in maintaining CTMT temperature and pressure, rather than cooling the supply air to these components. "A" and "B" are plausible, since CDS supplies the systems that provide CRDM cooling and RCP motor cooling.

Technical Reference(s): P&IDs 121B (Rev. 21), 122A (Rev. 20), and 122B (Rev. 10)
(Attach if not previously provided, OP 3353.MB1C (Rev. 21), 4-8
including version/revision number.) OP 3353.MB1C (Rev. 21), 5-5, step 4.5
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure (partial or complete) of one or more of the Containment Ventilation Sub-systems, determine the effects on the system and on interrelated systems.
Question Source: Bank #402633
Question History: Last NRC Exam Millstone 3 2009 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4 and 41.5
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 46 | Tier # | 2 | 2 |
| K/A Statement: Physical connections / cause effect | Group # | 1 | 1 |
| relationship between ESFAS and CVCS | K/A # | 013K1.11 | |
| Proposed Question: | Importance Rating | 3.3 | 3.8 |

With the plant initially at 100% power, the following sequence of events occurs:

1. Safety Injection automatically actuates due to Low Steam Pressure.
2. The crew enters E-0, *Reactor Trip or Safety Injection*.
3. The RO reports that neither train of CIA has actuated.
4. The RO reports that RCS pressure has decreased from 2250 psia to 1950 psia.
5. NO operator actions have been taken.

What is the current position of the following CVCS System valves?

| | Charging Header Isolation Valves<br>3CHS*MV8105 / 8106 | Charging Cold Leg Injection Valves<br>3SIH*MV8801A/B |
|---|---|---|
| a) | CLOSED | OPEN |
| b) | CLOSED | CLOSED |
| c) | OPEN | OPEN |
| d) | OPEN | CLOSED |

Proposed Answer: B

Explanation (Optional):
"C" and "D" are wrong, since when SIS actuates, the Charging Isolation Valves automatically CLOSE.
"C" and "D" are plausible, since CIA failed to actuate, CIA closes several CVCS System valves, and the Charging Isolation Valves isolate a line that penetrates Containment.
"B" is correct and "A" wrong, since the cold leg injection valves are interlocked to remain CLOSED until RCS pressure drops less than P-19 setpoint of 1900 psia, which is below current RCS pressure.
"A" is plausible, since the Cold Leg Injection Valves receive an auto-open on an SIS and RCS pressure has dropped.

Technical Reference(s): P&ID 113A (Rev. 33)
(Attach if not previously provided, Functional Sheet 8 (Rev. K)
including version/revision number.) LSKs 26-2.3D (Rev. 13), 26-2.3E (Rev. 10) and 27-2.A (Rev. 11)
Proposed references to be provided to applicants during examination: None
Learning Objective: For the below listed plant events, partial or complete, describe the effects on the Chemical and Volume Control System … Safety Injection Actuation… Containment Isolation Signal Phase A
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 47 | Tier # | 2 | 2 |
| K/A Statement: Main Feedwater System: Effect of | Group # | 1 | 1 |
| a loss or malfunction of MFW on the SG's | K/A # | 059K3.03 | |
| Proposed Question: | Importance Rating | 3.5 | 3.7 |

The plant is initially at 100% power.

The controlling channel of Feed Flow for the "D" Steam Generator fails HIGH.

Assume NO operator actions are taken, and the plant responds as designed. How will "D" SG Narrow Range Level respond to this event after its initial shrink or swell?

a) Level begins to decrease due to the "D" Feed Reg Valve throttling closed. Level will stabilize prior to the Lo-Lo Level Reactor Trip, and slowly return to 50%.
b) Level begins to decrease due to the "D" Feed Reg Valve throttling closed. Level will stabilize prior to the Lo-Lo Level Reactor Trip, and then remain stable at about 40%.
c) Level begins to increase due to the "D" Feed Reg Valve throttling open. Level will stabilize prior to the Hi-Hi Level P-14 FWI, and slowly return to 50%.
d) Level begins to increase due to the "D" Feed Reg Valve throttling open. Level will stabilize prior to the Hi-Hi Level P-14 FWI, and then remain stable at about 60%.

Proposed Answer: A

Explanation (Optional):
The maximum feed flow input to SGWLC is 133%, so when the feed flow transmitter to the S/G failed high, it created a 33% flow error (100% steam flow, 133% feed flow) for the affected S/G.
"C" and "D" are wrong, since with sensed feed flow greater than steam flow, a signal is sent to the FRV to throttle in the close direction. Also, there a lag circuit in the level error signal to delay its input during the effects of shrink or swell. "C" and "D" are plausible, since if the level error were the first to respond, it would cause level to initially start increasing. Also, if steam flow had failed high, this would be the initial response.
"A" is correct, and "B" wrong, since the Feed Reg Valve having throttled closed, and S/G level decreasing, a level error signal will result that will call for the FRV to throttle open. A thumb-rule for flow error to level error is 3.3% level error to compensate for 10% flow error. So a flow error of 33% will take an initial level error of about 10% to offset the flow error. This will cause SG level to initially stabilize when level reaches about 40%. Since the SGWLC level error signal integrates over time (a level dominant system), level will eventually be restored to 50%, even with the flow error present.
"B" is plausible, since the initial level where flow error and level error offset each other is 40%, and this would be correct if the level error signal did not integrate.

Technical Reference(s): Functional Sheet 13 (Rev. K)
(Attach if not previously provided, Functional Sheet 14 (Rev. K)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given the following failures (partial or complete) of the Main Feedwater & Steam Generator Water Level Control Systems, DETERMINE the effects on the system & on interrelated systems…

Question Source: Bank #405151
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5 and 41.7
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 48 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of the process used to track inoperable alarms | Group # | 2 | 2 |
| | K/A # | G2.2.43 | |
| Proposed Question: | Importance Rating | 3.0 | 3.3 |

A crew desires to disable a recurring nuisance annunciator.

Complete the following statements related to this process.

The crew will track the disabled annunciator by filling out a "Disabled Annunciator Data Sheet" found in __(1)__; and flag the disabled annunciator by placing a ____(2)____ on the annunciator window.

| | (1) | (2) |
|---|---|---|
| a) | OP-AA-1500, *Operational Configuration Control* | green triangle |
| b) | OP-AA-1500, *Operational Configuration Control* | blue dot |
| c) | OP 3353, *Annunciator System* | green triangle |
| d) | OP 3353, *Annunciator System* | blue dot |

Proposed Answer: D

Explanation (Optional):

Per OP 3353, Section 4.3, "Disabling Annunciators", if a nuisance alarm exists, the SM/US can have the annunciator placed in SILENCE.

"D" is correct, since the crew is then required to fill out Attachment 2, "Disabled Annunciator Data Sheet", and then place the appropriate color dot on the annunciator window. Per the notes prior to step 4.3.1, for an annunciator that is removed from service or is a nuisance alarm, a blue dot is attached.

"A" and "B" are wrong, since OP-AA-1500 doesn't provide guidance on disabling annunciators.

"A" and "B" are plausible, since OP-AA-1500 provides guidance to shift operators for maintaining proper control of plant configuration.

"C" is wrong, since the green triangle is placed on an annunciator to indicate that the annunciator receives its inputs from a local panel. "C" is plausible, since a green triangle is used on annunciator windows.

Technical Reference(s): OP 3353 (Rev. 09), Section 2.3, and Att. 2
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: As outlined in OP 3353, Annunciator System, describe the process used to disable and track inoperable annunciators.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 49 | Tier # | 2 | 2 |
| K/A Statement: Spent Fuel Pool Cooling System: Predict | Group # | 2 | 2 |
| impact / mitigate a loss of the Spent Fuel Pool Cooling System | K/A # | 033A2.02 | |
| Proposed Question: | Importance Rating | 2.7 | 3.0 |

The plant is at 100% power with initial conditions as follows:

- The "A" Spent Fuel Pool Cooling Pump is in service in its normal lineup.
- The "B" Spent Fuel Pool Cooling Pump breaker is racked out and tagged out for Breaker preventative maintenance.

The following sequence of events occurs:

1. The FUEL POOL TEMP HI annunciator is received on MB1.
2. The RO reports Fuel Pool temperature indicates 125°F.
3. The crew enters EOP 3505A, *Loss of Spent Fuel Pool Cooling*.
4. A PEO reports there is no RPCCW flow to the "A" Spent Fuel Pool Cooler.
5. The PEO reports RPCCW Supply Valve to "A" Spent Fuel Pool Cooler (3CCP*V110) is vibrating excessively, and that a stem-disk separation is likely to have occurred.
6. The RO stops the "A" Spent Fuel Pool Cooling Pump.
7. The PEO reports all other local SFC valve positions are in their normal positions.
8. The crew realigns RPCCW to the "B" Spent Fuel Pool Cooler.

What is the quickest procedurally directed method the crew can take to align the SFC System to cool the spent fuel pool using the "B" Spent Fuel Pool Cooler?

a) Rack in the "B" SFC Cooling Pump Breaker, align the "B" Spent Fuel Pool Cooler for service, and start the "B" SFC Cooling Pump.
b) Close the "A" Spent Fuel Pool Cooler Outlet Valve, throttle open the "B" Spent Fuel Pool Cooler Outlet Valve, and start the "A" SFC Cooling Pump.
c) Align temporary power from the "A" Train to the "B" SFC Cooling Pump, align the "B" Spent Fuel Pool Cooler for service, and start the "B" SFC Cooling Pump.
d) Open the "A" SFC Cooling Pump suction and discharge cross-connect valves, throttle open the "B" Spent Fuel Pool Cooler Outlet Valve, and start the "A" SFC Cooling Pump.

Proposed Answer: B

Explanation (Optional):
With Fuel Pool temperature high, EOP 3505A will direct the crew to restore SFP cooling per Att. B, "Restore Spent Fuel Pool Cooling". This has the crew stop the running SFC Pump, check one SFC Pump aligned (the "A" Pump is aligned), and check one SFP Cooler aligned (The "B" SFP Cooler is available), by checking (or aligning as needed) a SFP Cooler inlet valve open and outlet valve throttled open.
"B" is correct, since aligning an opposite train SFP Cooler to an available SFC pump requires stopping the operating SFC Pump, closing the same-train cooler outlet valve, throttling open the opposite train cooler outlet valve, and starting the desired SFP Cooling Pump.
"A" and "C" are wrong, since specific guidance is provided in OP 3305 to aligning an opposite train SFP Cooler to an available SFP Cooling pump.
"A" is plausible, since if the cooler is train specific to its train's pump (CCI Pump Cooling), and the "A" train cannot be restored, this would be the quickest way to restore SFP Cooling.
"C" is plausible, since in MODEs 5, 6, and 0, during train outages, temporary power is made available to power one train's pump from the opposite train power supply.
"D" is wrong, since the SFC pumps discharge into a common header before being routed to either of two parallel coolers. Normally, the running pump is aligned to the same train cooler.

“D” is plausible, since several cooling systems have cross connect valves to allow aligning a pump to the opposite train (SWP, RPCCW, TPCCW, and CDS).

Technical Reference(s): OP 3353.MB1A (Rev. 8), 4-4
(Attach if not previously provided, EOP 3505A (Rev. 15), Attachment D, step 5
including version/revision number.) OP 3305 (Rev. 28), Section 4.3.3
P&ID 111A (Rev. 39)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major action categories within EOP 3505A, "Loss of Spent Fuel Pool Cooling".

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4 and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 50 | Tier # | 2 | 2 |
| K/A Statement: Main Turbine Generator System: Predict | Group # | 2 | 2 |
| impact / mitigate malfunction of Electrohydraulic Control | K/A # | 045A2.17 | |
| Proposed Question: | Importance Rating | 2.7 | 2.9 |

The plant is at 100% power with turbine load control on "Load Set". The following sequence of events occurs:

1. C-16 comes in, and the EHC "HOLD" light illuminates.
2. The crew diagnoses a failed input to the HOLD circuit.
3. The crew successfully removes the failed input signal to the HOLD circuit.
4. The crew is preparing to restore from HOLD operation.

Complete the following statement about what input failure could have caused C-16 to actuate, and how the crew will restore from HOLD operation.

__(1)__ channel failed low. The crew will __(2)__ to clear the HOLD light.

| | (1) | (2) |
|---|---|---|
| a) | A Narrow Range $T_{hot}$ | lower the Load Limit pot |
| b) | A Narrow Range $T_{hot}$ | depress the HOLD pushbutton |
| c) | The selected Turbine Impulse Pressure | lower the Load Limit pot |
| d) | The selected Turbine Impulse Pressure | depress the HOLD pushbutton |

Proposed Answer: A

Explanation (Optional):
C-16 interlock stops turbine loading if:
• Auctioneered Tave decreases to 553°F OR
• Tave minus Tref = -20°F (Tave 20° below the Tref signal) ("C" and "D" wrong).
"A" is correct, since if Thot fails low, it will drop more than 20°F below Tref, and the C-16 HOLD condition is cleared by coming off of Load Set control. This is accomplished by lowering the Load Limit pot until it takes control away from Load Set.
"B" is wrong, since depressing the HOLD pushbutton will not remove turbine control from Load Set.
"B" is plausible, since this instrument failure will bring in C-16, and numerous pushbuttons on the EHC insert on MB7 switch between two operating modes.
"C" and "D are wrong, since if PT-505/506 fails low, Tref will go to approximately 557°F, but this will cause Tave - Tref to be a positive value (+30°F) while C-16 requires Tave - Tref to be a negative value.
"C" and "D" are plausible, since Turbine Impulse Pressure inputs to the C-16 circuit, and this failure would create a large mismatch between Tave and Tref.

Technical Reference(s): OP 3323A (Rev. 25), Section 4.14
(Attach if not previously provided, Functional Sheet #9 (Rev. H)
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure, partial or complete, of the EHC Control System, determine the effects on the system and on inter-related systems.
Question Source: Bank #403248
Question History: Last NRC Exam Millstone 3 2004 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3, 41.5, and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 51 | Tier # | 1 | 1 |
| K/A Statement: Loss of DC Power: Determine / interpret that a loss of dc power has occurred; verify substitute sources have come on line | Group # | 1 | 1 |
| | K/A # | APE058.AA2.01 | |
| | Importance Rating | 3.7 | 4.1 |

Proposed Question:

Initial conditions:

- The plant is at 100% power.
- DC Bus 4 is being supplied by its normal charger.

An earthquake occurs, resulting in the following sequence of events:

1. MCC 32-2W deenergizes, resulting in the expected annunciators on MB 8.
2. The BOP operator reports the BATTERY 4 TROUBLE annunciator is lit on MB 8.
3. A PEO reports the Battery 4 Output Breaker has tripped open.
4. A PEO is dispatched to re-close the Battery Output Breaker, and then place Swing Charger 8 in service on DC Bus 4.

Complete the following statement about what DC Bus 4 voltage indicates on MB8.

Before any PEO actions are taken, voltage indicates __(1)___volts. After the swing charger is placed in service, voltage indicates ___(2)___volts.

| | (1) | (2) |
|---|---|---|
| a) | 0 | 135 |
| b) | 0 | 140 |
| c) | 125 | 135 |
| d) | 125 | 140 |

Proposed Answer: A

Explanation (Optional):
Normally, Vital Instrument AC Bus VIAC-4 is supplied by a rectifier, which converts 480VAC to 140 VDC, which is supplied to the inverter. The inverter converts 140 VDC to 120VAC, which is supplied to the VIAC. This is backed up by the DC bus, which normally is supplied by a charger that puts out 135 VDC. This is backed up by the battery, which puts out 125 VDC.
"C" and "D are wrong, since in this event, when 480V MCC 32-2W is lost, the normal 480 Volt supply to the rectifier is lost, and power has been lost to Battery Charger 4, which is the normal charger for DC Bus 4. Also, the battery output breaker is open, so the DC Bus has lost all inputs, so the DC Bus is deenergized, and it will indicate 0 volts. "C" and "D" are plausible, since normal battery voltage is 125VDC.
"A" is correct, and "B" wrong, since when swing charger 8 is placed in service (powered from 480 VAC MCC 32-2U), the DC bus will be energized by the swing charger, which puts out 135 volts.
"B" and "D" are plausible, since normal rectifier output voltage is 140 Volts. Also, the output of the charger while on an equalizing charge is 140 Volts.

Technical Reference(s): EE 1BA (Rev. 31)
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure of the 125 VDC distribution system or a portion of the system, DETERMINE the effects on the system and on interrelated systems…

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7, 41.8, and 41.10

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 52 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of general guidelines | Group # | 4 | 4 |
| for EOP usage | K/A # | G2.4.14 | |
| Proposed Question: | Importance Rating | 3.8 | 4.5 |

Complete the following statement about the rules for EOP step completion.

While progressing through the EOP network, the crew is generally________.

a) allowed to proceed to the next step before completing the current step. If the steps must be completed before proceeding, a CAUTION prior to step 1 of the EOP will explicitly state the requirement
b) allowed to proceed to the next step before completing the current step. If a step must be completed before proceeding, either the logic or condition of the step, a WHEN/THEN statement, or a CAUTION will express the requirement
c) required to complete the current step before proceeding to the next step. If a step need not be fully completed before proceeding to the next step, a CAUTION prior to step 1 of the EOP will explicitly state the allowance
d) required to complete the current step before proceeding to the next step. If a step need not be fully completed before proceeding, either the logic or condition of the step, a WHEN/THEN statement, or a CAUTION will express the allowance

Proposed Answer: B

Explanation (Optional):
"C" and "D" are wrong, since unless otherwise specified, a step need not be fully completed before proceeding to the next step. Once a step is begun, the SM/US may determine it desirable and acceptable to continue the procedure actions even though the current task is not yet complete; however, completing the task in a timely manner is still required.
"C" and "D" are plausible, since in System Operating Procedures (OPs) and General Operating Procedures (GOPs), the crew is generally required to complete the current step before proceeding on to the next step.
"B" is correct, and "A" wrong, since if a particular step or portion of an EOP is required to be completed prior to proceeding to the next step, one of the following methods are used to alert the Operator of this situation:

- A "CAUTION" is provided explicitly stating this requirement.
- An RNO is provided with a WHEN, THEN logic statement which prohibits proceeding to the next step until the condition is satisfied.
- A specific condition in the step requires a hold until a required condition is satisfied.
- The logic of the step prevents the Operator from proceeding until a specific condition is satisfied.

"A" is plausible, for GOPs, a Caution or Note is used when the crew is allowed to proceed to the next step before completing the current step.

Technical Reference(s): OP 3272 (Rev. 12) Section 3.4, pages 14 and 15
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe when actions of a step need not be fully completed prior to proceeding to the next step within the same procedure or transitioning to another procedure (in EOPs and AOPs)

Question Source: Bank #408082
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 53 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of radiation exposure limits under normal or emergency conditions | Group # | 3 | 3 |
| | K/A # | G2.3.4 | |
| Proposed Question: | Importance Rating | 3.2 | 3.7 |

Current Conditions:

- A Site Area Emergency has been declared due to a LOCA outside CTMT.
- The LOCA is into the ESF building and a pathway to the environment exists.
- Limited makeup to the RWST is available.
- An operator is sent in to the ESF building to locally isolate the leak.
- The Assistant Director, Technical Support has approved an emergency exposure upgrade for the operator.
- This action will result in a significant reduction in offsite dose, protecting a large population.

The operator's current TEDE dose for the current year is 200 mrem.

What is the maximum emergency exposure this operator may receive while performing this action?

a) 4300 mrem TEDE
b) 4500 mrem TEDE
c) 24800 mrem TEDE
d) 25000 mrem TEDE

Proposed Answer: D

Explanation (Optional):
"A" and "B" are wrong, since emergency exposure limits for lifesaving or protection of large populations is 25 rem. "A" is plausible, since this dose would bring the worker's annual dose to 4.5 rem if the worker's previous dose for the year were included, and 4.5 rem is the maximum TEDE dose allowed for this event without dose extension. "B" is plausible, since 4.5 rem is the maximum TEDE dose allowed for this event without dose extension.
"D" is correct, and "C" wrong, since the emergency limit is independent of previous dose. "C" is plausible, since this dose would bring the worker's dose to 25 rem for the year, if the worker's previous dose for the current year was required to be included.

Technical Reference(s): MP-26-EPI-FAP09 (Rev. 002), Attachment 3
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: State the radiation exposure guidelines which have been established for emergencies and the considerations for applying those guidelines.

Question Source: Bank #369971
Question History: Last NRC Exam Millstone 3 2009 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.12 and 43.4
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 54 | Tier # | 1 | 1 |
| K/A Statement: Loss of Main Feedwater: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for EOPs and AOPs | Group # | 1 | 1 |
| | K/A # | APE054G.2.4.4 | |
| | Importance Rating | 4.5 | 4.7 |

Proposed Question:

With the plant initially at 100% power, the following sequence of events occurs:

1. The BOP announces that the 5A Feedwater heater level is rising
2. The 5A Feedwater heater HI HI level is reached
3. The BOP observes the following annunciators are lit:

- MB6A 1-4 "Feedwater Heater Level Hi-Hi"
- MB6A 1-6 "Condensate Demin System Trouble"
- MB6A 2-4 "Feedwater Heater Level Hi"
- MB6A 2-7 "Condensate Demin DP Hi"
- MB6A 5-7 "Htr Drn Pp Auto Trip Overcurrent"

What is the crew's top priority to address this event?

a) Enter AOP 3567, *Operation with One Feedwater Heater String Isolated,* and isolate extraction steam to the 5A Feedwater heater.

b) Enter AOP 3567, *Operation with One Feedwater Heater String Isolated,* and start the standby Condensate Pump.

c) Use ARP MB6A 5-7, "Htr Drn Pp Auto Trip Overcurrent", and reduce reactor power to less than 93.4% power.

d) Use ARP MB6A 2-7, "Condensate Demin DP Hi", and throttle 3CNM-MOV78, Condensate Demineralizer Bypass, to reduce differential pressure to less than 70 psid.

Proposed Answer: B

Explanation (Optional):

"B" is correct, since a tube leak has occurred on the 5th or 6th Pt Feedwater Heater (as determined by both a Hi-Hi level and auto trip of the associated Heater Drain Pump). In this situation, the low pressure feed string will auto isolate and the undersized bypass line will auto open. The bypass line is undersized and will result in inadequate condensate flow to the suction of the Main Feed Pumps. As a result, both operating Main Feed Pumps will trip on low suction pressure (within a few minutes from event initiation) if the BOP is slow to respond. As such, the crew needs to diagnose this event and enter AOP 3567. AOP 3567 will direct starting the standby Condensate Pump; thereby preventing a Reactor trip.

"A" is wrong since AOP 3567 prioritizes starting a third condensate pump. "A" is plausible as AOP 3567 will be used to isolate extraction steam (after the standby Condensate Pump is started).

"C" is wrong since ARP MB6A 5-7 would check HDP Pp parameters first and then direct going to AOP 3567 (if a feedheater string is isolated). "C" is plausible as this ARP is used to address a trip of a HDL Pump (without a Feedwater Heater String isolated) and it will direct a downpower.

"D" is wrong since ARP MB6A 1-4 would direct throttling 3CNM-MOV78 to less than 65 psid (not 70 psid). "D" is plausible, as this ARP would help with the Feed Pump suction pressure issue described above.

Technical Reference(s): AOP 3567 (Rev. 8), Steps 1 & 2,
(Attach if not previously provided, OP 3353.MB6A (Rev. 18), 1-4, 5-7
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning

Objective: IDENTIFY plant conditions which require entry into AOP 3567
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.7, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 55 | Tier # | 2 | 2 |
| K/A Statement: Auxiliary Feedwater: Operational implications of relationship between AFW flow and RCS heat transfer | Group # | 1 | 1 |
| | K/A # | 061K5.01 | |
| | Importance Rating | 3.6 | 3.9 |

Proposed Question:

With the plant initially at 100% power, the reactor trips.

The crew has just entered ES-0.1, *Reactor Trip Response*, and current conditions as follows:

- RCS pressure: 2200 psia and slowly lowering
- RCS Tave: 552ºF and slowly lowering
- PZR level: 25% and slowly lowering
- SG pressures: 1020 psig and slowly lowering
- All RCPs: Running
- AFW Flow: 300 gpm per SG
- No operator actions have been taken.

What is the first action the crew is required to take per ES-0.1 to address the cooldown?

a) Close the MSIV's and MSIV bypass valves.
b) Commence immediate boration.
c) Throttle Auxiliary Feedwater flow.
d) Initiate SI and return to step 1 of E-0.

Proposed Answer: C

Explanation (Optional):
"C" is correct, since decay heat initially drops rapidly on a reactor trip, and shortly after the trip, 1200 gpm of AFW flow will be more than enough to remove decay heat, and an RCS cooldown will occur. So ES-0.1 step 1.b directs reducing AFW to 530 to 600 gpm.
"A" wrong, since closing MSIVs will only be required if RCS temperature does not stabilize after throttling AFW flow. "A" is plausible, since closing MSIVs and Bypass Valves will be required in step 1.h RNO if RCS Temperature is still lowering after other means of stopping the cooldown have not been successful.
"B" wrong, since immediate boration will only be required if RCS temperature lowers to less than 530°F. "B" is plausible, since immediate boration will be required in step 1.h RNO if the cooldown causes Tave to lower to < 530ºF.
"D" is wrong, since foldout page SI requirements have not been met (PZR level < 9%, or subcooling < 32°F). "D" is plausible, since due to the cooldown, Pzr level is decreasing toward the SI actuation foldout page setpoint. SI actuation requirements will eventually be met on low Pzr level if the cooldown is not stopped.

Technical Reference(s): ES-0.1 (Rev. 30), steps 1.a-h
(Attach if not previously provided, OP 3272 (Rev. 12), Attachment 2, section 2.0, page 2 of 13
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: State the conditions which would allow the action of either the throttling or isolation of auxiliary feed water flow to a Steam Generator

Question Source: Bank #408732
Question History: Last NRC Exam Millstone 3 2004 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.5, 41.8, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 56 | Tier # | 1 | 1 |
| K/A Statement: Steam Line Rupture – Excessive Heat | Group # | 1 | 1 |
| Transfer: Operational implications of EOP warnings, cautions, and notes. | K/A # | APE040G.2.4.20 | |
| | Importance Rating | 3.8 | 4.3 |

Proposed Question:

The crew has entered ECA-2.1, *Uncontrolled Depressurization of All Steam Generators*.

The crew is in the process of terminating Safety Injection when "B" SG pressure begins to increase.

What action is required to be taken by the crew?

a) Remain in ECA-2.1 until the procedure has been completed, then transition to ES-1.1, *SI Termination.*
b) Suspend performance of ECA-2.1, and transition to ES-1.1, *SI Termination.*
c) Suspend performance of ECA-2.1, and transition to E-2, *Faulted Steam Generator Isolation*.
d) Complete SI termination steps in ECA-2.1, then transition to E-2, *Faulted Steam Generator Isolation.*

Proposed Answer: D

Explanation (Optional):
The crew has entered ECA-2.1 since there are no intact Steam Generators. If the crew is successful at regaining an intact SG, they will generally immediately transition back to E-2. Per a Caution prior to the SI termination steps, the exception to this transition requirement is if the crew is terminating SIS.
"D" is correct, and "A", "B", and "C" wrong, since the crew has commenced terminating SIS, so they are required to complete the SIS termination steps in ECA-2.1, and then transition to E-2 prior to completing all of the steps in ECA-2.1.
"A" is plausible, since ECA-2.1will take actions to terminate SI, and ES-1.1 will provide further restoration steps after SIS is terminated.
"B" is plausible, since immediate transition would be appropriate if SI termination steps were not in progress, and ES-1. 1 will terminate SIS and provide further restoration steps after SIS is terminated.
"C" is plausible, since immediate transition would be appropriate if SI termination steps were not in progress, and E-2 was the procedure in effect when the crew transitioned to ECA-2.1.

Technical Reference(s): ECA-2.1 (Rev. 21), CAUTION prior to step 11
(Attach if not previously provided, ECA-2.1 (Rev. 21), Foldout Page item 3
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of ECA-2.1
Question Source: Bank #407709
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 57 | Tier # | 1 | 1 |
| K/A Statement: LOCA Outside Ctmt: Determine / interpret adherence to appropriate procedures | Group # | 1 | 1 |
| | K/A # | WE04EA2.2 | |
| Proposed Question: | Importance Rating | 3.6 | 4.2 |

A LOCA outside Containment occurs, and the crew completes all steps of ECA-1.2, *LOCA outside Containment*.

A LOCA into which of these heat exchangers will **NOT** be addressed by the actions in ECA-1.2?

a) RCP "A" Thermal Barrier Heat Exchanger
b) CVCS System Letdown Heat Exchanger
c) Seal Water Return Heat Exchanger
d) "A" RHR Heat Exchanger

Proposed Answer: A

Explanation (Optional):
"A" is correct, since ECA-1.2 does not attempt to isolate the RPCCW System path to or from the RCP thermal barrier heat exchangers.
"B" is wrong, since ECA-1.2 step 2 checks the valve lineup for the charging and letdown paths from the RCS. This would isolate the Letdown line, which would isolate the Letdown Heat Exchanger
"C" is wrong, since ECA-1.2 step 2 checks the valve lineup for the charging and letdown paths from the RCS. This would isolate the Seal Return line, which would isolate the Seal Water Heat Exchanger.
"B" is wrong, since Step 3 isolates the RHR cold leg injection valves, which would isolate the "A" RHR Heat Exchanger.
"B", "C", and "D" are plausible, since all four Heat Exchangers have Reactor Coolant flowing through them.

Technical Reference(s): ECA-1.2 (Rev. 10), steps 2, 3, and 4
(Attach if not previously provided, P&ID 103A (Rev. 29), 104A (Rev. 54), 112A (Rev. 50)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major action categories contain within EOP 35 ECA-1.2

Question Source: Bank #407681
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3, 41.8, and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 58 | Tier # | 3 | 3 |
| K/A Statement: Ability to control radiation releases | Group # | 3 | 3 |
| | K/A # | G2.3.11 | |
| Proposed Question: | Importance Rating | 3.8 | 4.3 |

With the plant at 100% power and a SLCRS Fan running, the "SLCRS Filter Fan Discharge to Millstone Stack" radiation monitor (3HVR*RE19B) goes into alarm.

In accordance with AOP 3573, *Radiation Monitor Alarm Response*, what are two subsequent actions that are required to be performed to check for potential sources of the radioactive release?

A. Check the Condenser Air Ejector Discharge Radiation Monitor 3ARC21 trend, and check the CTMT Purge Exhaust Fans running.
B. Check the RPCCW Radiation Monitor 3CCP31 trend, and check the CTMT Purge Exhaust Fans running.
C. Check the Condenser Air Ejector Discharge Radiation Monitor 3ARC21 trend, and check the CTMT Vacuum Pumps running.
D. Check the RPCCW Radiation Monitor 3CCP31 trend, and check the CTMT Vacuum Pumps running.

Proposed Answer: C

Explanation (Optional):
"C" is correct since the Condenser Air Ejectors and the CTMT Vacuum Pumps are both potential sources of the radiation release that will be checked per AOP 3573, since they both discharge to Gaseous Waste, which exhausts to the Millstone Stack.
"A" and "B" are wrong, since AOP 3573 will not direct the crew to check the Containment Purge System, which exhausts to the Turbine Bldg stack.
"A" and "B" are plausible, since the CTMT Purge System draws on CTMT, as do the CTMT vacuum pumps.
"B" and "D" are wrong, since, AOP 3573 will not direct the crew to check the RPCCW rad monitor trend.
"B" and "D" are plausible, since the bulk of the RPCCW system is in the Aux Bldg, which is drawn on by SLCRS, and the RPCCW system is monitored by CCP-RE31.

Technical Reference(s): AOP 3573 (Rev. 28), Attachment A, Page 11
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories within AOP 3573.
Question Source: Bank #407254
Question History: Last NRC Exam Millstone 3 2017 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.8, 41.10, 41.11, and 43.4
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 59 | Tier # | 2 | 2 |
| K/A Statement: Containment Cooling System: Effect a loss or malfunction of Containment Cooling will have on Containment Instrumentation readings | Group # | 1 | 1 |
| | K/A # | 022K3.02 | |
| | Importance Rating | 3.0 | 3.3 |

Proposed Question:

The plant is at 100% power when the following sequence of events occurs:

1. One of the two running CAR Fans trips.
2. The standby CAR Fan will not start.
3. Containment temperature starts to increase.
4. No operator actions have been taken.

Complete the following statement about how Pressurizer level indication 3RCS*LI459 will initially change, and whether 3RCS*LI459 will indicate higher or lower than Cold Cal Pzr level indication 3RCS-LI462.

LI459 indication will initially __(1)__, and indicate __(2)__ than cold calibrated level indicator LI462.

| | (1) | (2) |
|---|---|---|
| a) | decrease | lower |
| b) | decrease | higher |
| c) | increase | lower |
| d) | increase | higher |

Proposed Answer: D

Explanation (Optional):
"A" and "B" are wrong, since a temperature increase in the general area will heatup the reference leg, causing the reference leg water to become less dense. This will cause LT-459 to indicate higher than actual level because the reference leg mass has decreased. "A" and "B" are plausible, since this would be true if the variable leg heated up.
"D" is correct, and "C" wrong, since the cold cal instrument (LT462) is calibrated for 140°, so the water it senses in the Pressurizer is less dense than that for which it is calibrated, so it reads artificially low. Since LT459 is calibrated for a higher temperature it initially indicates higher than LT462, and will always read higher than the Cold Calibrated channel, even though the reference leg temperature is changing.
"C" is plausible, since the cold cal instrument is calibrated to a different temperature than the hot cal instruments, and conditions are changing.

| Technical Reference(s): (Attach if not previously provided, including version/revision number.) | OP 3208 (Rev. 34), Attachment 2 |
|---|---|
| | OP 3208 (Rev. 34), Note prior to step 4.2.12 |
| | P&ID 102C (Rev. 27) |
| | GFS |

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure, partial or complete, of the Pressurizer Pressure and Level Control System, determine the effects on the system and on interrelated systems.

Question Source: Bank #404218

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3 and 41.7

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 60 | Tier # | 1 | 1 |
| K/A Statement: Loss of Intermediate Range NIS: | Group # | 2 | 2 |
| Determine / interpret loss of compensating voltage | K/A # | APE033AA2.11 | |
| Proposed Question: | Importance Rating | 3.1 | 3.4 |

A reactor shutdown is in progress per OP 3207, *Reactor Shutdown*, and the following sequence of events occurs:

1. The RO commences inserting the Shutdown Bank Rods.
2. Several minutes later, the RO reports the following IRNI indications:

| | Power Level | Startup Rate |
|---|---|---|
| • IR channel N-35: | $4.0 \times 10^{-9}$ amps | 0.0 DPM |
| • IR channel N-36: | $1.1 \times 10^{-11}$ amps | 0.0 DPM |

What IRNI compensation issue exists, and how is this affecting the plant?

a) Channel N-35 is undercompensated. This is preventing both the Source Ranges from automatically energizing.
b) Channel N-35 is overcompensated. This is preventing both the Source Ranges from automatically energizing.
c) Channel N-36 is undercompensated. This caused both Source Ranges to energize prematurely, potentially causing an unwarranted reactor trip.
d) Channel N-36 is overcompensated. This caused both Source Ranges to energize prematurely, potentially causing an unwarranted reactor trip.

Proposed Answer: A

Explanation (Optional):
During a reactor shutdown, the expected SUR is -1/3 dpm, until $10^{-11}$ Amps is reached, where a trickle current maintains indication at $10^{-11}$ Amps. At this point, 0 dpm SUR will be indicated.
"C" and "D" are wrong, since neither channel shows -1/3 DPM, which means power has actually entered the Source Range. And since IR channel N-35 indicates above $10^{-11}$ amps, it is the channel with the compensation issue. Since channel N-35 indicates high, not enough gamma current is being removed, which means it is undercompensated. "C" and "D" are plausible, since with both channels showing 0 DPM SUR as opposed to the expected -1/3 DPM SUR, it is not immediately apparent which channel has the problem. Also, if both channels are overcompensated, IRNIs could drop below P-6 early, with SRNIS above 105 cps, resulting in a SR High Flux Trip.
"A" is correct, and "B" wrong, since N-35 indicating high prevents P-6 (2 of 2 channels $<10^{-10}$ amps) from automatically energizing the source ranges. AOP 3571, Attachment E, step 2 states that if an IR fails during a shutdown OR has symptoms of undercompensation, actuate both source range resets when reactor power is believed to be in or near the Source Range.
"B" is plausible, since overcompensation could imply indicating current is too high.

Technical Reference(s): AOP 3571 (Rev. 17), Attachment E, steps E.13, E.14, and E.15
(Attach if not previously provided, Functional Sheets 3 (Rev. G) & 4 (Rev. G)
including version/revision number.) GFS

Proposed references to be provided to applicants during examination: None

Learning Objective: For the following conditions, determine the effects on the Nuclear Instrumentation System and on interrelated systems… Under-compensated Intermediate Range detector…

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 61 | Tier # | 2 | 2 |
| K/A Statement: Emergency Core Cooling System: Ability to locate and operate components, including local controls | Group # | 1 | 1 |
| | K/A # | 006G2.1.30 | |
| Proposed Question: | Importance Rating | 4.4 | 4.0 |

A loss of all Service Water (SWP) Pumps occurs, resulting in the following sequence of events:

1. The crew enters AOP 3560, *Loss of Service Water*.
2. NO Service Water Pumps can be started.
3. The RO trips the Reactor.
4. The crew enters E-0, *Reactor Trip or Safety Injection*.
5. A PEO completes AOP 3560, Attachment B "Establishing Alternate Charging Pump Cooling" for the running "B" Charging Pump.

Now that Attachment B has been completed, cooling is via feed and bleed cooling, with ___(1)___ being supplied to the ___(2)___ side of the CCE Heat Exchanger.

| | (1) | (2) |
|---|---|---|
| a) | Fire Water | SWP |
| b) | Fire Water | CCE |
| c) | RPCCW | SWP |
| d) | RPCCW | CCE |

Proposed Answer: A

Explanation (Optional):
This question is considered a KA match since Millstone 3 utilizes the Charging Pumps as the High Head Safety Injection Pumps in the event of an accident. AOP 3560 directs using Attachments "A", "B", and "D" if a transition to E-0 is made. Attachment "A" has RCP trip criteria, Attachment "B" aligns feed and bleed cooling to the Charging Pumps, which is required since Charging Pump cooling is normally supplied by Service Water, which has been lost. Attachment "D" aligns alternate SI Pump cooling.
"A" is correct, and "B", "C", and "D" wrong, since Attachment "B" aligns Fire Water to the Service Water Side of the CCE Heat Exchanger.
"B", "C", and "D" are plausible, since RPCCW normally supplies makeup to the CCE Heat Exchanger, and supplying feed and bleed cooling to either side of the heat exchanger would cool the Charging Pump. Also, when aligning alternate cooling to the SI Pumps, Domestic Water is aligned to the fresh water side of the Heat Exchanger.

Technical Reference(s): AOP 3560 (Rev. 13), Att. B, step B.3
(Attach if not previously provided, including version/revision number.) AOP 3560 (Rev. 13), Step 2.a.RNO

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major action categories contained within AOP-3560

Question Source: Bank #406931
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.8 and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 62 | Tier # | 2 | 2 |
| K/A Statement: Station Air System: Design feature / interlock which provides for cross-connect with Instrument Air | Group # | 2 | 2 |
| | K/A # | 079K4.01 | |
| Proposed Question: | Importance Rating | 2.9 | 3.2 |

With the plant initially at 100% power, the following sequence of events occurs:

1. The RO reports that Instrument Air (IAS) header pressure is at 88 psig and decreasing.
2. The crew enters AOP 3562, *Loss of Instrument Air.*
3. A PEO reports an air leak on an IAS branch line in the turbine building.
4. Service Air to Instrument Air Cross-Connect Valve 3IAS-AOV14 automatically opens, and IAS pressure stabilizes.
5. A PEO has just been dispatched to isolate the leak.

Complete the following statement:

The current position of Service Air Supply Valve 3SAS-AOV33 is __(1)__, and if needed after the leak is isolated, the PEO can locally close Cross-Connect Valve 3IAS-AOV14 by rotating its control switch at __(2)__.

| | (1) | (2) |
|---|---|---|
| a) | OPEN | IAS annunciator panel 3IAS-PNLIS |
| b) | OPEN | Service Air Compressor gauge board 3SAS-PNL2 |
| c) | CLOSED | IAS annunciator panel 3IAS-PNLIS |
| d) | CLOSED | Service Air Compressor gauge board 3SAS-PNL2 |

Proposed Answer: C

Explanation (Optional):
Pressure switch 3IAS-PS14 senses IAS common header pressure downstream of the IAS receivers.
"A" and "B" are wrong, since PS14 will cause AOV14 to open and AOV33 to close when pressure lowers to 85 psig. "A" and "B" are plausible, since when cross connect valve 3IAS-AOV14 automatically opens Service Air will supply Instrument Air whether or not AOV33 closes, and if AOV 33 were to remain open, Service Air pressure would not be lost.
"C" is correct, and "D" wrong, since the valves can be locally aligned via switches at the IAS annunciator panel. "D" is plausible, since the valves are aligning the Service Air Compressor discharge path, and there are controls for the Service Air compressor at the Service Air Compressor gauge board.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
OP 3353.IS (Rev. 03), 1-1
LSK-12-1C (Rev. 6)
LSK-12-2C (Rev. 8)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following plant air systems components controls and interlocks… Service air to instrument air cross-connect valve (IAS-AOV14)…

Question Source: Bank #395503
Question History: Last NRC Exam Millstone 2 2011 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4 and 41.7
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 63 | Tier # | 2 | 2 |
| K/A Statement: Effect of loss or malfunction of the | Group # | 1 | 1 |
| AC distribution system will have on the EDG | K/A # | 062K3.02 | |
| Proposed Question: | Importance Rating | 4.1 | 4.4 |

With the plant initially at 100% power, the following sequence of events occurs:

1. VIAC-1 de-energizes.
2. On the resulting transient, Safety Injection actuates.
3. On the plant trip, offsite power is lost.

Complete the following statement about how the "A" EDG will automatically respond to this event prior to operator action.

The "A" EDG___(1)___start, and Bus 34C___(2)___strip

| | (1) | (2) |
|---|---|---|
| a) | WILL | WILL |
| b) | WILL | WILL NOT |
| c) | WILL NOT | WILL |
| d) | WILL NOT | WILL NOT |

Proposed Answer: B

Explanation (Optional):
Normally on a SIS plus an LOP, the EDG automatically starts, the bus strips, the EDG output breaker closes, and loads sequence on. On a loss of VIAC-1 the "A" EDG sequencer deenergizes. The loss of a sequencer with SIS actuating results in the following:
1. The associated diesel will not start
2. Loads will not be stripped from the emergency bus.
3. Load sequencing will not occur on the associated bus.

On an LOP, an EDG start signal will also be generated directly from LOP relays from the 4KV Emergency bus.
"C" and "D" wrong since the EDG starts on a LOP signal from the emergency bus. "C" and "D" are plausible, since on a loss of the sequencer on a SIS without a LOP, the EDG would not start.
"B" is correct, and "A" wrong, since the bus does not strip. "A" is plausible, since on a SIS with LOP this is what would normally occur.

Technical Reference(s): AOP 3564 (Rev. 13), Caution prior to step 1
(Attach if not previously provided, LSK 24-9.4A (Rev. 12)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, properly apply the notes and cautions of AOP 3564.

Question Source: Bank #407018
Question History: Last NRC Exam Millstone 3 2011 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7 and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 64 | Tier # | 1 | 1 |
| K/A Statement: ATWS: Ability to operate / monitor manual rod control | Group # | 1 | 1 |
| | K/A # | EPE029.EA1.09 | |
| Proposed Question: | Importance Rating | 4.0 | 3.6 |

With the plant initially at 100% power, the following sequence of events occurs:

1. The Turbine trips.
2. The Reactor fails to trip.

Current conditions are as follows:

- Reactor power is 85% and decreasing.
- Tave is 593°F and increasing.
- The crew is entering FR-S.1, *Response to Nuclear Power Generation/ATWS.*

Assuming there is a delay in opening the Reactor Trip Breakers locally, what is the procedurally directed sequence for the RO to add negative reactivity to the reactor as quickly as possible?

a) Leave Rod Control in AUTOMATIC until all rods are on the bottom, monitoring that rods are driving in as expected.
b) Initially leave Rod Control in AUTOMATIC, and then place rods in MANUAL and drive rods in when that becomes the faster way to insert Control Rods.
c) Initially place Rod Control in MANUAL and drive rods in, and then place Rod Control back to AUTOMATIC when that becomes the faster way to insert Control Rods.
d) Initially place Rod Control in MANUAL and drive rods in. Keep driving rods in in MANUAL until all rods are on the bottom.

Proposed Answer: B

Explanation (Optional):

FR-S.1, step 1 RNO directs the operators to insert Control Rods in Auto or Manual. This provides the operators the flexibility to use whichever method causes rods to insert the fastest. Rod speed in Manual is 48 steps per minute. Rod speed in Automatic varies between 0 and 72 steps per minute, based on temperature error and rate of power change difference between the reactor and the turbine.
"C" and "D" are wrong, since initially, rods will drive in faster in Automatic, since current temperature error exceeds 5°F, and the turbine trip created a large reactor to turbine power rate of change signal.
"B" is correct, and "A" wrong, since as rods drive in in automatic, temperature will lower, and the power mismatch signal will decay away, so rod speed will begin to slow down in Automatic; and when it becomes faster to drive rods in in Manual, the RO will switch to manual rod control and drive rods in.
"A", "B", and "D" are plausible, since FR-S.1 allows the use of Auto or Manual rods, and rods will be driven in manually and automatically at some point.

Technical Reference(s): FR-S.1 (Rev. 23), step 1 RNO
(Attach if not previously provided, including version/revision number.) ROD001 Lesson Plan (R7C1), Section A.6 (page 9 and 10 of 69)

Proposed references to be provided to applicants during examination: None

Learning Objective: Prioritize the Operator-initiated recovery techniques that would mitigate the consequences of an ATWS.

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5, 41.6, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 65 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of facility requirements | Group # | 1 | 1 |
| for controlling vital / controlled access | K/A # | G2.1.13 | |
| Proposed Question: | Importance Rating | 2.5 | 3.2 |

Two Unit 3 Operators have been assigned to escort 10 visiting people while giving them a tour of the Transformer and Switchgear areas of Unit-3. The following conditions exist:

- The only vital area that the visitors have been given authorization for entry is the switchgear area.
- The tour is progressing from the transformer area in the yard to the East Switchgear Room.
- Prior to entering the switchgear room, one of the escorts is paged, and is required to return to the control room.

What action is in accordance with SC-1, *Access and Egress Control*?

A. Both escorts are required to remain with the visitors and escort them outside the protected area prior to the one escort leaving for the control room.
B. The one remaining escort may take escort responsibility for all 10 visitors and remain outside of the switchgear room until the second escort returns.
C. The one remaining escort may take 5 visitors into the switchgear area, while the other escort takes the 5 remaining visitors into the control room.
D. The one remaining escort may take escort responsibility for all 10 visitors and continue the tour into the switchgear area.

Proposed Answer: B

Explanation (Optional):
An escort is required to maintain both observation and control of visitors. Escort/ visitor ratios are 10 to 1 for non-vital areas and 5 to 1 for vital areas.
"B" is correct, and "A" wrong, since as long as the tour has not entered a vital area, one escort for 10 visitors is acceptable. "A" is plausible, since an escort needs to leave the tour, and this would be correct if the vital area ratio was applicable to the entire protected area.
"C" is wrong, since the visitors are not authorized to enter the control room. "C" is plausible, since 5 to 1 ratio is acceptable for vital areas.
"D" is wrong, since the switchgear is a vital area, with a 5/1 rule. "D" is plausible, since 10 to 1 is acceptable in the protected area.

Technical Reference(s): SC-1 (Rev. 011-02), Sections 1.8.5.c and 1.14.2
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe escorting responsibilities.

Question Source: Bank #369939
Question History: Last NRC Exam Millstone 3 2009 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 66 | Tier # | 1 | 1 |
| K/A Statement: Fuel Handling Incidents: Operational implications of indications of approaching criticality | Group # | 2 | 2 |
| | K/A # | APE036AK1.03 | |
| Proposed Question: | Importance Rating | 4.0 | 4.3 |

Initial conditions:

- The plant is in MODE 6.
- A core re-load is in progress per OP 3310B, *Refueling Operations*.

Over the course of a few moments, the following sequence of events occurs:

1. The SHUTDOWN MARGIN MONITOR CHANNEL 1 (MB4C 2-2) Annunciator is received.
2. Both of the Source Range channels indicate an unexplained flux increase.
3. The SHUTDOWN MARGIN MONITOR CHANNEL 2 (MB4C 2-3) Annunciator is received.

What are two actions the crew is required to take?

a) Immediately suspend CORE ALTERATIONS, and enter EOP 3502, *Fuel Handling Accident*.
b) Initiate Control Building Isolation (CBI), and enter EOP 3502, *Fuel Handling Accident*.
c) Immediately suspend CORE ALTERATIONS and enter AOP 3566, *Immediate Boration*.
d) Initiate Control Building Isolation (CBI), and enter AOP 3566, *Immediate Boration*.

Proposed Answer: C

Explanation (Optional):
"C" is correct, since a Shutdown Margin Monitor alarm is based on Gamma-metrics Source Range count increasing indicating a positive reactivity event is in progress, and if left unchecked, an inadvertent criticality may occur. Per OP 3310B, the crew is required to Immediately suspend core alterations and place the fuel in a stable condition, evacuate Containment, have HP check out all personnel exiting Containment, and enter AOP 3566, *Immediate Boration*.
"A" and "B" are wrong, since entry into EOP 3502 is required based on receiving a report from the fuel handling operators, or from high radiation conditions, neither of which are present.
"A" and "B" are plausible, since fuel handling operations are in progress, and this may be related to the loss of Shutdown Margin.
"D" is wrong, since initiating CBI is not required per OP 3210B or AOP 3566.
"D" is plausible, since actuating CBI will protect the Control Room staff if a radiation release is in progress. Also, if the crew enters EOP 3502, the crew will be directed to actuate CBI.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
OP 3210B (Rev. 13), Step 4.2.4
AOP 3566 (Rev. 15), Sections 2.1 and 2.2
OP 3353.MB4C (Rev. 22), 2-2
EOP 3502 (Rev. 10), Section 2 and step 6

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss conditions requiring transition to other procedures from OP 3210 series

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5, 41.10, and 41.15
Comments:

| | | | | |
|---|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 67 | Tier # | 1 | 1 |
| K/A Statement: Loss of Emergency Coolant Recirculation: Operational implications of components, capacity, and function of emergency systems | Group # | 1 | 1 |
| | K/A # | WE11EK1.1 | |
| | Importance Rating | 3.7 | 4.0 |

Proposed Question:

A LOCA is in progress, and current conditions are as follows:

- The crew is in ECA-1.1, *Loss of Emergency Coolant Recirculation*.
- RCS pressure is stable at 700 psia.
- The crew has just verified that ECCS is in service.

Prior to checking if ECCS flow can be terminated, what is the optimal ECCS lineup the crew will establish per ECA-1.1 for core heat removal?

a) “A” Train CHS Pump and “B” SIH Pump running only
b) “A” Train CHS Pump, “B” SIH Pump, and “A” or “B” RHR Pump running only
c) Two Charging Pumps and two SIH Pumps running only
d) Two Charging Pumps, two SIH Pumps, and two RHR Pumps running

Proposed Answer: A

Explanation (Optional):
The strategies of ECA-1.1 are to restore recirc capability, delay depletion of the RWST, and cooldown and depressurize the RCS to RHR conditions. ECCS capacity ensures a single train of High Head Injection (one CHS and one SIH Pump) is adequate to remove decay heat. To delay the depletion of the RWST, the procedure directs the crew to makeup to the RWST and to minimize RWST depletion by stopping unnecessary containment spray pumps and decreasing the SI pump flowrate.
“A” is correct, and “B”, “C”, and “D” wrong, since ECA 1.1 directs the crew to reduce SI flow to one Charging and one SIH Pump with RCS pressure above 300 psia. The NOTE before step 23 cautions the operators to stop High Head Injection pumps on alternate trains.
“B” is plausible, since this is the minimum ECCS Pump alignment if RCS pressure is below 300 psia.
“C” is plausible as this is normally the optimum lineup after successfully transferring to Cold Leg Recirc.
“D” is plausible, since this is normally the optimum lineup prior to reaching RWST Lo-Lo level setpoint.

Technical Reference(s): ECA-1.1 (Rev 19), Step 23, and associated NOTE.
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories within EOP 35 ECA-1.1
Question Source: Bank #407668
Question History: Last NRC Exam Millstone 3 2019 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.8 and 41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 68 | Tier # | 2 | 2 |
| K/A Statement: Monitor automatic operations of the | Group # | 1 | 1 |
| Emergency Diesel Generator, including the purpose of | K/A # | 064A3.12 | |
| the automatic load sequencer | Importance Rating | 3.3 | 3.5 |

Proposed Question:

Initial Conditions:

- The plant is at 100% power.
- The "A" Charging Pump is running.

A Loss of off-site power occurs, and both EDGs and both Sequencers respond as designed.

Complete the following statements concerning the response of the Charging Pumps to the LOP.

The "A" CHS Pump breaker__(1)__. After the EDGs re-energize emergency buses 34C and 34D, the__(2)__.

a) (1) remains closed
   (2) "A" CHS Pump restarts immediately, and the "B" CHS Pump breaker will close at time = 0 seconds
b) (1) remains closed
   (2) "A" CHS Pump restarts immediately, and the "B" CHS Pump will remain off
c) (1) opens on the sequencer bus strip signal
   (2) "A" and "B" CHS Pump breakers will both close at time = 0 seconds
d) (1) opens on the sequencer bus strip signal
   (2) "A" CHS Pump breaker will close at time = 0 seconds, and the "B" CHS Pump will remain off

Proposed Answer: B

Explanation (Optional):
The purpose of the Sequencer is to prevent overloading the EDGs when they energize the bus by stripping off large loads, and sequencing the loads on after various time delays based on their importance to the event in progress.
"C" and "D" are wrong, since when a LOP occurs, all 4KV emergency bus loads are stripped except the Load Centers and Charging pumps. This ensure the previously running ("A") CHS Pump to be the first large load to start after the EDGs energize the buses.
"C" and "D" are plausible, since all 4KV emergency bus loads other than the CHS Pumps are stripped.
"B" is correct and "A" wrong, since only one CHS Pump is required on a LOP. "A" is plausible, since if the LOP is combined with any other event, such as SIS, the second charging pump will be sequenced on at T=0 by the sequencer.

Technical Reference(s): LSK 24-9.4A (Rev. 12)
(Attach if not previously provided, P&ID 104A (Rev. 54)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the emergency diesel load sequencers under the following normal, abnormal, and emergency conditions: Automatic operation: LOP only…

Question Source: Bank #403194

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.8

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 69 | Tier # | 1 | 1 |
| K/A Statement: Small Break LOCA: Determine / interpret whether Pzr water inventory loss is imminent | Group # | 1 | 1 |
| | K/A # | EPE009EA2.06 | |
| Proposed Question: | Importance Rating | 3.8 | 4.3 |

The reactor has tripped with a RCS leak in progress, and initial conditions are as follows:

- "B" Train 4KV buses 34B and 34D are deenergized.
- The crew has entered ES-0.1, *Reactor Trip Response*.
- Charging Flow is at the maximum rate, with Control Valve 3CHS*FCV121 in MANUAL.
- Pzr Level is stable at 20%.

The following sequence of events occurs:

1. RCS pressure and Pzr level start decreasing.
2. The STA estimates the RCS leakrate exceeds current charging capability by 40 gpm.

Assuming no operator actions are taken, about how long would it take for Pzr level to decrease from its initial level to reach the ES-0.1 Foldout Page SI Actuation criterion?

a) Between 0 and 4 minutes
b) Between 6 and 10 minutes
c) Between 12 and 16 minutes
d) Between 18 and 22 minutes

Proposed Answer: D

Explanation (Optional):
ES-0.1 foldout page criteria for SI actuation has the crew actuate SI and Go to E-0 if RCS subcooling based on CETC is less than 32 °F OR Pressurizer level less than 9%.
"D" is correct, and "A", B", and "C" wrong, since initial Pzr level was at 20%; and SI actuation criterion is 9%, so 20% - 9% = 11% decrease in level required. The volume in each % of Pzr level above 2200 psia is about is about 75 gallons / %. So 11% x 75 gallons/% = 825 gallons. The leak rate given was 40 gpm, so 825 gallons / 40 gallons/minute = 20.6 minutes to reach 9% Pzr level.
"A", "B", and "C" are plausible, since depending on the assumed Pzr volume / % level, and the assumed SIS foldout page criterion (procedures where a RCS depressurization has occurred have a foldout page criterion of 16% rather than 9%), values within these ranges could be obtained.

Technical Reference(s): ES-0.1 (Rev. 30), Foldout Page, Item 1
(Attach if not previously provided, SP3601F.6-001 (Rev. 04), Calculation Table 1A
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of ES-0.1.

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3, 41.5, and 41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 70 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of fire protection procedures | Group # | 4 | 4 |
| | K/A # | G2.4.25 | |
| Proposed Question: | Importance Rating | 3.3 | 3.7 |

With the plant at 100% power, the following sequence of events occurs:

1. Fire alarms are received for the “B” EDG at the Color Graphics Unit in the Control Room.
2. A PEO reports smoke coming from the “B” EDG Enclosure.

What actions, if any, are required to be taken by the crew per OP 3341A, *Fire Protection Water System* to initiate Fire Protection Water (FPW) flow to the “B” EDG?

a) No operator action is required. When individual wet pipe sprinkler heads melt, flow from that sprinkler head will initiate.
b) Open the deluge valve manually from the Color Graphics Unit to initiate flow through all of the open-head deluge sprinkler heads.
c) Open the manual sprinkler cutout valve locally in the “B” EDG west vestibule. Then, when the sprinkler heads melt, flow will initiate.
d) Open the manual deluge valve locally in the “B” EDG west vestibule to initiate flow through all of the open-head deluge sprinkler heads.

Proposed Answer: C

Explanation (Optional):
“C” is correct, and “A”, “B”, and “D” wrong, since the EDGs are protected by a manual, pre-action closed sprinkler system, with its manual isolation valve kept normally closed. The UV detectors warn of a fire, and to initiate sprinkler flow, the manual sprinkler cutout valve is manually opened. This valve must be opened via a remote reach rod in the west vestibule to pressurize the system, and the sprinkler heads must melt before flow will occur.
“A” is plausible, since the automatic, wet type sprinkler system is the most widely used sprinkler system at Millstone 3. Also, this would be correct if the EDG manual isolation valve were normally kept open.
“B” is plausible, since open head deluge sprinkle systems are used in several locations at Millstone 3. To initiate flow, a clapper valve must open. The clapper valve latch automatically opens when heat detectors sense a fire. The Color Graphics Unit has the capability to actuate the deluge clapper valves manually from the Control Room, but this feature is currently blocked.
“D” is plausible, since opening a manual valve is required to initiate flow to the EDG, and deluge fire water systems are used at several locations at Millstone 3.

Technical Reference(s): OP 3341A (Rev. 18-0), Section 1.2, pages 3-6
(Attach if not previously provided, OP 3341A (Rev. 18-0), Section 4.16
including version/revision number.) P&ID 146D (Rev. 27)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the following Water Fire Protection (FPW) sub-systems including the areas within unit-3 which are protected by each… Manual, pre-action, closed head sprinkler system…
Question Source: Bank #403428
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 71 | Tier # | 3 | 3 |
| K/A Statement: Knowledge of "Fire in the Plant" procedures | Group # | 4 | 4 |
| | K/A # | G2.4.27 | |
| Proposed Question: | Importance Rating | 3.4 | 3.9 |

With the plant initially at 100% power, the following sequence of events occurs:

| Time | Event |
|---|---|
| 0700: | Two fire alarms are received at the Color Graphics Unit in the Control Room for the Auxiliary Building, 24' south floor area. |
| 0701: | A PEO calls the Control Room and reports smoke in the Auxiliary Building, 24' level. |
| 0702: | The crew enters EOP 3509, *Fire Emergency*. |

When was/is the crew first required to go to the specific fire area procedure (EOP 3509.2, *Aux. Bldg. El. 24' 6", South Floor Area, 43' 6" & 66' 6" Fire*)?

a) When the two separate fire alarms were received at the Color Graphics Unit
b) When the PEO reported smoke in the Auxiliary Building, 24' level.
c) When the crew entered EOP 3509, *Fire Emergency*.
d) When EOP 3509 directs the crew to go to EOP 3509.2.

Proposed Answer: D

Explanation (Optional):
One entry condition for EOP 3509 is "Alarms on the control room fire console." EOP 3509, step 3.e stops the crew at this step until fire or smoke is verified present in the area. Per the RNO, the crew exits the procedure if fire or smoke is not present.
"D" is correct, and "A", "B", and "C" wrong, since the transition from EOP 3509 to the fire procedure for the specific area does not occur until EOP 3509, step 7, which directs the crew to the EOP for the affected area while continuing to take actions in 3509. Also, the entry conditions for 3509.2 states that this procedure is entered from EOP 3509, Fire Emergency "when a determination of fire in this area has been made."
"A" is plausible, since two fire alarms were received, increasing the likelihood that an actual fire is present.
"B" is plausible, since smoke was reported in the area, providing visual indication that a fire is present.
"C" is plausible, since EOP 3509 was entered, and actions from the two procedures will be performed in parallel after the crew enters the fire procedure for the specific area.

Technical Reference(s): EOP 3509 (Rev. 027-0) Entry Conditions
(Attach if not previously provided, EOP 3509 (Rev. 027-0), steps 3.e, 5, and 7
including version/revision number.) EOP 3509.2 (Rev. 008-0) Entry Conditions
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories contained within the EOP 3509 series procedures
Question Source: Bank #407842
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 72 | Tier # | 2 | 2 |
| K/A Statement: Emergency Diesel Generator System: | Group # | 1 | 1 |
| Manually operate / monitor adjustment of exciter voltage | K/A # | 064A4.02 | |
| Proposed Question: | Importance Rating | 3.3 | 3.4 |

The BOP operator is performing a Test-Start of the "A" EDG per OP 3346A, *Emergency Diesel Generator*, and is ready to parallel the Diesel to Bus 34C.

"A" EDG parameters indicate as follows:

- "INCOMING" voltage: 4000 Volts
- "RUNNING" voltage: 4150 Volts

Complete the following statement about how the BOP operator is required to adjust voltage in accordance with OP 3346A.

The BOP will place the VOLTAGE REGULATOR CONTROL Switch in the ___(1)___ position to adjust voltage until incoming voltage is slightly ___(2)___ than running voltage.

| | (1) | (2) |
|---|---|---|
| a) | "RAISE" | greater |
| b) | "RAISE" | less |
| c) | "LOWER" | greater |
| d) | "LOWER" | less |

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" are wrong, since going to "RAISE" is required to raise incoming (EDG) voltage until it is slightly greater than running (Bus) voltage.
"B" is plausible, since the initial voltage difference requires raising incoming voltage regardless of whether the target is slightly less or slightly more than running voltage.
"C" is plausible, since if EDG voltage was considered the "running" voltage, voltage would be lowered to obtain incoming voltage slightly greater than running voltage.
"D" is plausible, since if EDG voltage was considered the "running" voltage, voltage would be lowered to obtain incoming voltage slightly less than running voltage.

Technical Reference(s): OP 3346A (Rev. 46), step 4.6.1.g
(Attach if not previously provided, GFS
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the emergency diesel generator system under the following… Parallel Operation

Question Source: Modified Bank #403147 (Parent Question included below)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:
This question is considered “modified” since incoming voltage was lowered in the stem to make it plausible that the crew would have to raise voltage to obtain incoming voltage slightly less than running voltage. Also, the question was modified to ask the target voltage difference, rather than which switch is required to be used, making all four distractors different than in the original question.

Original Bank Question

The BOP operator is performing a Test-Start of the “A” EDG per OP 3346A, *Emergency Diesel Generator*, and is ready to parallel the diesel to Bus 34C.

“A” EDG parameters indicate as follows:

- “INCOMING” voltage: 4100 Volts
- “RUNNING” voltage: 4150 Volts

In accordance with OP 3346A, how is the BOP operator required to adjust EDG exciter voltage?

a) Place the “SPEED/LOAD” Switch in the “RAISE” position.

b) Place the “SPEED/LOAD” Switch in the “LOWER” position.

c) Place the “VOLTAGE REGULATOR CONTROL” Switch in the “RAISE” position.

d) Place the “VOLTAGE REGULATOR CONTROL” Switch in the “LOWER” position.

Answer: C

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 73 | Tier # | 2 | 2 |
| K/A Statement: Reactor Coolant Pumps: Physical connections / cause effect relationship between the RCPs and the Component Cooling Water System | Group # | 1 | 1 |
| | K/A # | 003K1.12 | |
| | Importance Rating | 3.0 | 3.3 |

Proposed Question:

With the plant at 100% power, the following sequence of events occurs:

1. A 20 gpm tube leak occurs in the “D” RCP thermal Barrier heat exchanger.
2. The RO monitors CCP flows on the Plant Process Computer.
3. The leak worsens, increasing to 100 gpm.
4. All systems respond as designed.
5. One minute later, with the plant still on line, the US directs the RO to again monitor RPCCW flows.

How will current CCP flows compare to the flowrates observed the first time the RO monitored flows?

a) “A” train RPCCW CTMT header flow has decreased.
b) “A” train RPCCW CTMT header flow has increased.
c) “B” train RPCCW CTMT header flow has decreased.
d) “B” train RPCCW CTMT header flow has increased.

Proposed Answer: A

Explanation (Optional):
A RCP thermal Barrier heat exchanger tube leak results in RCS leakage into the RPCCW system. To protect the RPCCW system, when flow reaches 86 gpm, the thermal barrier HX return isolation valve (3CCP*AOV178D) auto-closes to isolate the leak, isolating RPCCW flow through the "D" RCP thermal barrier.
"A" is correct, and “B” wrong, since isolating the thermal barrier line reduces RPCCW flow through the train "A" CTMT header. “B” is plausible, since the leak has worsened, and if this auto-isolation feature did not exist, of if its setpoint were greater than 100 gpm, RPCCW flow would have increased through the CTMT header.
"C" and "D" are wrong, since the “D” RCP is supplied by the “A” Train of RPCCW. "C" and “D” are plausible, since for several train-specific systems, Train A is assigned components "A" and "C", and Train B is assigned components “B” and “D”.

Technical Reference(s): P&ID 121B (Rev. 21)
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following Reactor Plant Component Cooling System equipment controls and interlocks… Thermal Barrier Return Isolation Valves…

Question Source: Bank #402426

Question History: Last NRC Exam Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3, 41.4, and 41.7

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 74 | Tier # | 2 | 2 |
| K/A Statement: Reactor Protection System: Bus power | Group # | 1 | 1 |
| supplies to RPS channels, components, and interconnections | K/A # | 012K2.01 | |
| Proposed Question: | Importance Rating | 3.3 | 3.7 |

What provides electrical power to the Train A (Orange Train) Solid State Protection System (SSPS) 48 Volt and 15 Volt Power Supplies?

a) VIAC 1 and 2 each provides a set of 48 Volt and 15 Volt Power Supplies.
b) VIAC 1 and 3 each provides a set of 48 Volt and 15 Volt Power Supplies.
c) VIAC 1 provides the 48 Volt Power Supply, and VIAC 2 provides the 15 Volt Power Supply.
d) VIAC 1 provides the 48 Volt Power Supply, and VIAC 3 provides the 15 Volt Power Supply.

Proposed Answer: B

Explanation (Optional): Each of the two Trains of SSPS receives power from the two train-related VIACs. "A" and "C" are wrong, since Train A receives power from VIAC 1 and VIAC 3. Each of the two trains of SSPS contains four Protection Sets, with each Protection Set powered by its own VIAC. "A" and "C" are plausible, since often, component 1 and 2 would align with Trains "A" and "B". "B" is correct and "D" wrong, since each VIAC provides redundant power to a 48V power supply and to a 15 Volt power supply. "D" is plausible, since VIAC 1 and 3 provide inputs to Train A SSPS that are not redundant, and VIAC 1 also provides power to the Train "A" SSPS Slave Relays for continuity testing.

Technical Reference(s): AOP 3564 (Rev. 13), Entry Conditions
(Attach if not previously provided, SSPS Power Distribution RPS Training Figure 5 (Rev 05-0)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basic power supply arrangement for control and protection channels including the specific channel's color scheme. (As available)

Question Source: Bank #410066

Question History: Millstone 3 2017 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 75 | Tier # | 2 | 2 |
| K/A Statement: Pressurizer Pressure Control System: Electrical power supply to the Controller for the Pzr Spray Valve | Group # | 1 | 1 |
| | K/A # | 010K2.02 | |
| Proposed Question: | Importance Rating | 2.5 | 2.7 |

The crew has just entered AOP 3563, *Loss of DC Bus*.

On a loss of which one of the following DC Buses will the Pzr Spray Valves still function?

a) DC Bus 1
b) DC Bus 2
c) DC Bus 4
d) DC Bus 5

Proposed Answer: C

Explanation (Optional):
"C" is correct, since a loss of Battery Bus 4 does not impact the use of Pzr Spray.
"A", "B", and "D" are wrong, since on a loss of either DC Bus 1, DC Bus 2, or DC Bus 5, AOP 3563 informs the crew that normal Pzr spray is NOT available, and if required to reduce RCS pressure, to use a Pzr PORV."
"A", "B", and "D" are plausible, since each of these are DC Buses at Millstone 3, and DC power is used to supply control power throughout the plant.

Technical Reference(s): AOP 3563 (Rev. 15), steps A1.5, B1.5, D1, and E1.6
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure of the 125 VDC distribution system or a portion of the system, DETERMINE the effects on the system and on interrelated systems…
Loss of DC bus effect on control systems…

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 76 | Tier # | | 1 |
| K/A Statement: Loss of Nuclear Service Water: | Group # | | 1 |
| Determine / interpret the valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition | K/A # | APE062AA2.03 | |
| | Importance Rating | | 2.9 |

Proposed Question:

The plant is in MODE 5, and initial conditions are as follows:

- An "A" Electrical Train Outage is in progress.
- The "B" Service Water Pump is running.

The following sequence of events occurs:

1. The RPCCW HX SW FLOW HI/LO annunciator is received on MB1C.
2. A PEO is dispatched, and reports a SWP pipe break just downstream of the "B" Train Service Water to RPCCW Supply Valve 3SWP*MOV50B.
3. The "B" Service Water Pump trips.
4. The RO isolates the break by closing 3SWP*MOV50B.
5. The crew is preparing to start the "D" SWP Pump.

With 3SWP*MOV50B isolated, what available Service Water Heat Exchanger(s) is/are required to be aligned to provide SWP minimum flow requirements, **and** prevent exceeding maximum SWP flow limits?

a) Two TPCCW Heat Exchangers only
b) One TPCCW Heat Exchanger only
c) The “C” RPCCW and one TPCCW Heat Exchanger
d) The “C” RPCCW Heat Exchanger only

Proposed Answer: A

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions while performing actions from an ARP, and apply information from the Service Water System procedure to mitigate the event in progress.
In order to provide minimum flow protection for the “A” train of Service Water with one pump running, either one RPCCW HX or two TPCCW HX’s is/are required to be aligned. In order to prevent exceeding the maximum flow limit for the “A” train of Service Water with one pump running, a maximum of one RPCCW HX and one TPCCW HX are allowed to be aligned. "A" is correct, since two TPCCW Heat Exchangers provide flow between the minimum and maximum requirements, and TPCCW has not been isolated.
“B” is wrong, since one TPCCW HX does not provide enough flow to meet the minimum flow requirement. "B" is plausible, since TPCCW provides a flowpath for SWP with 3SWP*MOV50B closed, and there is a concern with excessive flow as well as minimum flow.
"A" and "C" are wrong because closing 3SWP*MOV50B isolates the "B" AND the "C" RPCCW HX, so the “C” RPCCW Heat Exchanger is not available. "A" and "C" are plausible, since these HX alignments would be between the minimum and maximum flow limits, and detailed knowledge of the Service Water to RPCCW piping is required to determine isolating SWP to the "B" HX also isolates Service Water to the "C" HX.

Technical Reference(s): OP 3326 (Rev. 37), Precautions 3.9 and 3.10
(Attach if not previously provided, including version/revision number.) P&ID s 133A (Rev. 44), and 133B (Rev. 91)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Service Water System, and the basis for each.

Question Source: Bank #405272

Question History: Last NRC Exam Millstone 3 2015 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4 and 43.5

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 77 | Tier # | | 3 |
| K/A Statement: Knowledge of general operating crew responsibilities during emergency operations | Group # | | 4 |
| | K/A # | G2.4.12 | |
| Proposed Question: | Importance Rating | | 4.3 |

With the plant initially at 100% power, instrument air is lost, and the following sequence of events occurs:

1. The Reactor trips.
2. The crew enters ES-0.1, *Reactor Trip Response.*
3. The US desires to pull up an EOP step in order to restore Instrument Air prior to being directed to do so by ES-0.1.

In accordance with OP-AP-104, *Emergency and Abnormal Operating Procedures,* what administrative requirements are required to be carried out to pull this step forward?

a) Obtain concurrence from a second SRO and invoke 10CFR50.54X. Concurrence from the NRC is not required, but inform the NRC at the earliest opportunity.
b) Obtain concurrence from the Control Room crew, ensure EOP implementation is not delayed or interfered with, and use the specific EOP step.
c) Obtain Shift Manager permission, and document that a "variance" is being taken in the EOP.
d) Obtain Shift Manager permission, and document that a "deviation" is being taken in the EOP.

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level, since applicants are required to have knowledge of administrative procedures that specify coordination of plant emergency procedures.
“A” is wrong, since performing steps prior to the procedure prompt is allowed without invoking 10CFR50.54X. Actions to provide increased plant stability may be taken prior to reaching the procedural prompt. It is understood in some instances these actions are necessary to prevent further degradation of the plant or to protect the public or plant personnel.
“A” is plausible, since this is not normally done, and when done, it must be done with caution, since there are concerns with potentially masking symptoms of the event or defeating the intent of the emergency operating procedure being used. Also, the admin requirements specified are required if 10CFR50.54.X is invoked.
“B” is correct, since a step may be performed prior to the procedure prompt provided:
1. The control room team must not delay procedure implementation to overly diagnose plant conditions. If plant conditions are not clearly obvious to the team or the action necessary to control the event is not understood by the team, that action shall not be taken prior to the procedural prompt.
2. Steps performed early may not preclude or interfere with other actions required by the procedure.
3. Concurrence from control room team members is obtained prior to initiating the actions.
4. All required actions are performed using the applicable step(s) of the procedure.
“C” and “D” are wrong, since neither a variance nor a deviation is required to pull the step forward.
“C” and “D” are plausible, since control room crew concurrence is required, the SM is in charge of the overall performance of the crew while in EOPs, and a second licensed SRO’s permission is required when deviating from the EOPs (a step does not exist to adequately address the situation) per 10CFR50.54X.

Technical Reference(s): OP-AP-104 (Rev. 05), step 3.5.28.a
(Attach if not previously provided, OP-AP-104 (Rev. 05), Definitions 5.3.3. and 5.3.10
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe when actions of a step need not be fully completed prior to proceeding to the next step within the same procedure or transitioning to another procedure

Question Source: Bank #408083

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10 and 43.5

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 78 | Tier # | | 3 |
| K/A Statement: Knowledge of procedures and limitations | Group # | | 1 |
| involved in core alterations | K/A # | G2.1.36 | |
| Proposed Question: | Importance Rating | | 4.1 |

A refueling outage is in progress, and the crew is preparing to commence loading fuel into the core per OP 3210B, *Refueling Operations*.

Which of the following conditions would prevent the Refueling SRO from commencing fuel movement?

a) RHR flow rate from the running RHR Pump has been increased from 2,800 to 3,500 gpm.
b) RHR flow has deliberately been suspended for two hours (out of the last eight hours) to improve refueling cavity visibility.
c) The Containment Equipment Hatch is open under administrative control, and is capable of being closed and bolted in place with four bolts.
d) Audible NIS counts are being provided by Gammametrics rather than Westinghouse NIS.

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level since it requires the applicant to have knowledge of administrative requirements associated with refueling activities.
"A" is wrong, since RHR Pump flow has remained within the allowed range of 2,800 gpm to 4,000 gpm. "A" is plausible, since core alterations are in progress, and there is a required RHR Pump flow range.
"B" is correct, since the time RHR flow has been deliberately suspended exceeds the maximum time allowance of one hour in the last 8 hour period (if no dilution in progress).
"C" is wrong, since and the Containment Equipment Hatch is allowed to be open under administrative control. "C" is plausible, since the Equipment Hatch is open.
"D" is wrong, since Gammametrics detectors can suffice for Source Range Counts. "A" is plausible, since the normal method of monitoring counts during refueling is Westinghouse SR NIS.

Technical Reference(s): OP 3210B (Rev. 13), Prerequisites, especially 2.1.12 and 2.1.18
(Attach if not previously provided, OP 3210B (Rev. 13), Precautions, especially 3.9, 3.10, and 3.13
including version/revision number.) Tech Spec LCO 3.9.2 (Amendment 258)
SP3613F.3-001 (Rev. 06-01), page 2 of 2

Proposed references to be provided to applicants during examination: None

Learning Objective: Verify initial refueling requirements are met prior to movement of any fuel or core alterations

Question Source: Bank #403353
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10, 43.6, and 43.7
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 79 | Tier # | | 2 |
| K/A Statement: Engineered Safety Features Actuation System: | Group # | | 1 |
| Predict impact / mitigate LOCA | K/A # | 013.A2.01 | |
| Proposed Question: | Importance Rating | | 4.8 |

The crew has just transitioned to E-1, *Loss of Reactor or Secondary Coolant*, and current conditions are as follows:

- RCS pressure: 250 psia and stable.
- Total RHR pump flow: 1100 gpm per pump.
- CTMT temperature: 150°F.
- MCC 32-3T: Deenergized
- CDA Reset Pushbuttons on MB2: NOT functioning

Assuming these conditions do not change, what ES procedure is the crew required to transition to from E-1, and what GA will the crew be directed to implement immediately after entering the applicable ES procedure?

a) Enter ES-1.2, *Post LOCA Cooldown and Depressurization*, and use GA-1, *Energizing MCC-32-3T*.
b) Enter ES-1.2, *Post LOCA Cooldown and Depressurization*, and use GA-29, *Resetting ESF Actuation Signals Locally*.
c) Enter ES-1.3, *Transfer to Cold Leg Recirculation*, and use GA-1, *Energizing MCC-32-3T*.
d) Enter ES-1.3, *Transfer to Cold Leg Recirculation*, and use GA-29, *Resetting ESF Actuation Signals Locally*.

Proposed Answer: D

Explanation (Optional):
This question is considered SRO level, since it requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific sub-procedures. It also requires the ability to assess plant conditions and then select a GA procedure to mitigate an abnormal condition during an emergency.
"A" and "B" are wrong, since RCS pressure is below RHR shutoff head, and RHR Pump flow is greater than 1000 gpm, so a transition to ES-1.3 will be required.
"A" and "B" are plausible, since if either RCS pressure were above 300 psia, OR RHR Pump flow was below 1000 gpm, a transition to ES-1.2 would be required.
"D" is correct, since upon entering ES-1.3, the first action the crew will take is reset ESF actuation signals, and if this cannot be accomplished from the control room, the crew will be directed to use GA-29.
"C" is wrong, since GA-1 will not reenergize MCC 32-3T if CDA has actuated and offsite power is still available.
"C" is plausible, since just prior to the transition step in E-1, operators are directed to use GA-1 to reenergize MCC 32-3T. Also, maintaining power to the Plant Process Computer during complicated evolutions is helpful to the operators.

Technical Reference(s): E-1 (Rev. 27), Steps 8, 10.i, 12, and 14
(Attach if not previously provided, ES-1.3 (Rev. 20), Step 1.RNO
including version/revision number.) GA-1 (Rev. 05), Steps 1 and 2
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss conditions which require transition to other procedures from EOP 35 E-1
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7 and 43.5

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 80 | Tier # | | 2 |
| K/A Statement: Nuclear Instrumentation: Ability to | Group # | | 2 |
| apply Technical Specifications for a system | K/A # | 015G2.2.40 | |
| Proposed Question: | Importance Rating | | 4.7 |

The crew is performing a reactor startup in accordance with OP 3202, *Reactor Startup*, and initial conditions are as follows:

- The reactor is critical.
- The crew is preparing to block the NIS source ranges.
- NIS Indications are as follows:
  - SR Channel N-31: $3x10^4$ cps
  - SR Channel N-32: $5x10^4$ cps
  - IR Channel N-35: $7 x10^{-10}$ amps
  - IR Channel N-36: $3 x10^{-10}$ amps

Source Range NIS Channel N-32 fails low.

What ACTION, if any, is required to be taken by Technical Specifications? If none, what action is required/allowed OP 3202?

a) The startup may continue, but the inoperable channel must be restored to OPERABLE status within 48 hours or the crew must open the Reactor Trip System breakers within the next hour.
b) The startup may continue, but the inoperable channel must be restored to OPERABLE status within 7 days, or be in HOT SHUTDOWN within the next 12 hours.
c) No Tech Spec ACTION is currently required. Per OP 3202, suspend the startup, suspend all operations involving positive reactivity addition, and drive all control rods in.
d) No Tech Spec ACTION is currently required. Per OP 3202, block both SR channels and continue the startup.

Proposed Answer: D

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, determine whether a shutdown is required, and determine whether to remain in the current procedure or transition out of it. This decision goes beyond system knowledge and beyond knowing the overall strategy of the procedure. It also requires knowledge of Tech Spec applicability during an abnormal plant condition.
"D" is correct, and "A", "B", and "C" wrong, since the plant is in MODE 2 during a reactor startup with power above P-6, so both 3.3.1 and 3.3.3.5 are N/A. When the reactor is subsequently shut down, the plant will be required to enter an Action Statement, but continuing the startup keeps the plant in a condition where the failed NIS channel is not required.
"A" is plausible, since this action would be required per LCO 3.3.3.5 Action a if the reactor were shutdown, and the crew has not yet blocked the Source Ranges.
"B" is plausible, since this action would be required per LCO 3.3.1 Action 11 if the reactor were shutdown, and the crew has not yet blocked the Source Ranges.
"C" is plausible, since this action would be required per OP 3202 if the startup was in progress below P-6, and the crew has not yet blocked the Source Ranges.

Technical Reference(s): OP 3202 (Rev. 25), Step 4.35
(Attach if not previously provided, AOP 3571 (Rev. 17), Att. F, Step F.4.a.RNO
including version/revision number.) Tech Spec Table 2.2-1 (Amend. 242), FU 18.a
Tech Spec LCO 3.3.1, Action 11 (Amend. 266)
Tech Spec Table 3.3-1 (Amend. 266), FU 6.a, and applicable Mode 2##
Tech Spec LCO 3.3.3.5 (Amend. 258), Applicability and Action a

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a plant condition or equipment malfunction (with the NIS System)…
A. Determine entry conditions to applicable plant procedures
B. Evaluate Technical Specification Applicability and determine required actions

Question Source: Bank #406537
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.43.2 and 43.5
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 81 | Tier # | | 2 |
| K/A Statement: Containment Iodine Removal System: | Group # | | 2 |
| Predict impact / mitigate high temperature in the filter system | K/A # | 027.A2.01 | |
| Proposed Question: | Importance Rating | | 3.3 |

The plant is in MODE 5, at the start of a refueling outage, when the following initial sequence of events occurs:

1. In preparation for opening Containment for entry, the crew starts the "A" Containment Air Filtration Fan 3HVU-FN3A per OP 3313D, *Containment Air Filtration*.
2. The crew starts the "B" EDG per OP 3346A, *Emergency Diesel Generator*.
3. The crew shifts from the "A" to the "B" Containment Air Filtration Fan per OP 3313D.
4. Several minutes later, the following two annunciators are received at VP1:
   - CB INLET SMOKE
   - CTMT BLDG FLTR TEMP HI/HI-HI
5. PEOs report NO visible indication of smoke or flame locally at either location.

Which ARP is the highest priority for the crew to enter; and what action will the ARP direct the crew to take to mitigate this event?

a) CTMT BLDG FLTR TEMP HI/HI-HI. Open the "CTMT AIR FLTR X-TIE DMPR" (3HVU-AOD44) to provide cooling air flow.
b) CTMT BLDG FLTR TEMP HI/HI-HI. Start the second CAF Fan (3HVU-FN3A) to provide cooling air flow.
c) CB INLET SMOKE. Actuate Control Building Isolation (CBI) to protect the Control Building Filters.
d) CB INLET SMOKE. Align Control Building Ventilation to recirculation with outside filtered air to maintain breathable air in the Control Room.

Proposed Answer: A

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, and determine the appropriate ARP to enter based on abnormal conditions that go beyond system knowledge and beyond knowing the overall strategy of these ARPs. They also are required to determine what action to take per the ARP.
The Ctmt Filters are placed in service prior to Ctmt entry to reduce Ctmt radiation levels by removing radioactive nuclides from the Ctmt atmosphere. These radionuclides are absorbed in the Charcoal beds of the Ctmt filters, and high temperature in a Ctmt Building Filter is indicative of decay heat being generated from collected radionuclides. If this is excessive and not addressed, the potential exists for fire in the charcoal beds.
"A" is correct, since the concern with filter high temperature is excessive heat in the standby train (which is not being cooled by air flow), and this creates the potential for a fire in the charcoal filter. So the crew will open the cross-tie damper to allow cooling air flow through the standby filter.
"B" is wrong, since only one Containment Air Filtration System train is normally operated at any given time, unless a concerted effort is in process to clean up the containment atmosphere of radionuclides. "B" is plausible, since starting the second CAF fan would provide cooling to the filter. Also, if opening the cross-tie damper does not clear the alarm, or if the alarm reflashes (indicative or reaching the Hi-Hi temperature setpoint), the crew would be directed to stop the running CAF fan and start the standby CAF fan.
"C" and "D" are wrong, since CB inlet smoke ARP actions are not required unless actual fire or smoke is discovered. "C" and "D" are plausible, since actuating CBI, or aligning for recirculation with outside

filtered air, will pressurize the control room, preventing in-leakage from outside air into the Control Room, and the filters would filter contaminants already in the air.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)

OP 3353.VP1B (Rev. 10), 3-2, Corrective Actions 1.3, 1.4, and 3
OP 3353.VP1C (Rev. 14), 2-1A, Corrective Actions 2 and 3
OP 3313D (Rev. 08), Precautions 3.4 and 3.5
P&ID 153A (Rev. 29)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure (partial or complete) of one or more of the Containment Ventilation Sub-systems, determine the effects on the system and on interrelated systems.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.9 and 43.5

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 82 | Tier # | | 2 |
| K/A Statement: Reactor Coolant System: Predict impact / mitigate loss of forced circulation | Group # | | 2 |
| | K/A # | 002A2.03 | |
| Proposed Question: | Importance Rating | | 4.3 |

The crew is conducting a natural circulation cooldown per ES-0.4, *Natural Circulation Cooldown With Steam Void In Vessel (Without RVLMS)*, and current conditions are as follows:

- The crew is depressurizing the RCS while monitoring Pzr level.
- RCS Hot Leg temperature: 395°F & stable.
- RCS Pressure: 725 psia & decreasing.

The RO reports PZR level is 91% and increasing.

What action is the crew required to take based on PZR level?

a) Limit the cool down rate in the RCS cold legs to less than 50°F/hr.
b) Decrease RCS hot leg wide range temperatures to 350°F.
c) Increase RCS pressure by 100 psi using PZR heaters.
d) Maintain stable pressurizer level by reducing Charging flow.

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since applicants are required to assess plant conditions and determine the appropriate action to be taken based on abnormal conditions that go beyond system knowledge and beyond knowing the overall strategy of this event-specific Emergency Sub (ES) procedure. Based on plant conditions, it can be determined that the crew is depressurizing the RCS while looping though steps 16 and 17 of ES-0.4. The target pressure for the RCS is 650 psia, and the crew is maintaining RCS inventory stable by equalizing Charging and Letdown flow.

"A" is wrong, since the RCS cooldown rate is limited to 80°F/hr in ES-0.4. "A" is plausible, since 50°F/hr is the cooldown rate limit in ES-0.2 where a natural circulation cooldown is conducted while preventing void growth, but the crew has entered ES-0.4 due to a need to cooldown at a faster rate.

"B" is wrong, since PZR level has reached a depressurization termination criterion. This is met, so the crew is required to stop the depressurization. "B" is plausible, since this would be the correct action if Pzr level remained within the allowed band during the depressurization step.

"C" is correct, since PZR level is being used as an indirect indication for reactor vessel level (since RVLMS is not available). This is valid since as void size grows, RCS water moves up the surge line into the Pzr. And 91% Pzr level indicates that the void has grown to a point where the void is approaching the elevation of the RCS hot legs. Because of this, one depressurization termination criterion in step 16b is Pzr level >90%. This is met, so the crew is required to stop the depressurization, and step 17 directs the crew to repressurize the RCS to collapse the void and cool the head with cooler RCS water.

"D" is wrong, since charging and letdown are equalized during the depressurization steps in order to use Pzr level as an indirect indication of void size. "D" is plausible, since this action is required during the cooldown steps to make up for RCS shrink, to ensure Pzr level remains indicative of Vessel void size.

Technical Reference(s): ES-0.4 (Rev. 13-01), Steps 3, and 14 through 17
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per ES-0.4

| | | |
|---|---|---|
| Question Source: | Bank #408705 | |
| Question History: | Last NRC Exam | Millstone 3 2019 NRC Exam |
| Question Cognitive Level: | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.43.5 | |

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 83 | Tier # | | 1 |
| K/A Statement: Loss of Component Cooling Water: | Group # | | 1 |
| Knowledge of Limiting Conditions for Operations and safety limits | K/A # | APE026G.2.2.22 | |
| | Importance Rating | | 4.7 |

Proposed Question:

With the plant at 100% power, the following sequence of events occurs:

1. A PEO reports finding a manual valve on the "A" Train RPCCW System in the CLOSED position that should be OPEN.
2. The SM determines that the valve is in the wrong position, and that the valve supplies safety–related equipment.
3. The STA looks at the surveillance history of the RPCCW System, and discovers that the last scheduled monthly valve lineup surveillance (as specified in Surveillance Frequency Control Program) on the "B" Train RPCCW System was inadvertently missed.
4. The STA reports the missed "B" Train surveillance was scheduled for 6 days ago, and that the surveillance was last performed 36 days ago.
5. The crew has NOT repositioned the "A" Train valve to the required position.

**Using LCO/Surveillance Requirement 3/4.7.3 (RPCCW), attached to the back of this exam,** is the crew allowed to delay logging into a LCO ACTION STATEMENT? If yes, what is required with each Train? If no, what ACTION is required?

a) Yes. For Train A, the crew is allowed to delay logging into an ACTION for up to 24 hours while repositioning the valve.
For Train B, the crew is allowed to delay logging into an ACTION for up to 24 hours while completing the surveillance.

b) Yes. For Train A, the crew is not required to log into an ACTION as long as they immediately reposition the valve.
For Train B, the crew is allowed to delay logging into an ACTION for up to 2.75 days while completing the surveillance.

c) No. The crew is required to log into the ACTION for LCO 3.7.3 to restore "A" Train RPCCW to OPERABLE within 72 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d) No. The crew is required to log into the ACTION for LCO 3.0.3 initiate action within 1 hour to place the unit in HOT STANDBY within the next 6 hours, and HOT SHUTDOWN within the following 6 hours, and COLD SHUTDOWN within the following 24 hours.

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to apply the generic LCO and Surveillance requirements of section 3/4.0 of Tech Specs during abnormal plant conditions.
“A” and “B” are wrong, since the “A” Train valve is in the wrong position, and Surveillance Requirement 4.0.1 states failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LCO. “A” and “B” are plausible, since 4.0.2 allows a 25% time extension of the specified time interval to complete a surveillance (38.75 days for a monthly surveillance), and 4.0.3 allows 24 hours, or up to the limit of the specified surveillance interval, whichever is greater, to complete the surveillance if missed.
“C” is correct, since this is the ACTION required per LCO 3.7.3, and this is surveilled per 4.7.3 by verifying each valve servicing safety related equipment that is not locked, sealed, or otherwise secured in position is in its correct position.
“D” is wrong, since the valve affects the “A” Train only, so the LCO can be "met" by carrying out the ACTION STATEMENT. “D” is plausible, since this is the ACTION required if the LCO cannot be met by reliance on the ACTION STATEMENT, and a valve is out of position on the “A” Train, and a surveillance was inadvertently missed on the “B” Train.

| | |
|---|---|
| Technical Reference(s): | Tech Spec LCO 3/4.7.3 (Amendment 258) |
| (Attach if not previously provided, | Tech Spec LCO 3.0.3 and Surv. Req. 4.0.1 (Amendment 213) |
| including version/revision number.) | Tech Spec Surveillance Req 4.0.2 and 4.0.3 (Amendment 241) |

Proposed references to be provided to applicants during examination: **Tech Spec LCO/Surv Req 3/4.7.3**

Learning Objective: (For the Component Cooling Water System) Given a plant condition or equipment malfunction, use provided reference material to:
A. Determine entry conditions to applicable plant procedures
B. Evaluate Technical Specification applicability and determine required actions

| | |
|---|---|
| Question Source: | Bank #402471 |
| Question History: | Last NRC Exam Millstone 3 2007 NRC Exam |
| Question Cognitive Level: | Comprehension or Analysis |
| 10 CFR Part 55 Content: | 55.41.4 and 43.2 |
| Comments: | |

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 84 | Tier # | | 3 |
| K/A Statement: Ability to approve release permits | Group # | | 3 |
| | K/A # | G2.3.6 | |
| Proposed Question: | Importance Rating | | 3.8 |

With the plant at 100% power and all Radioactive Liquid Waste System Radiation Monitoring Instrumentation operating normally, the following sequence of events occurs:

1. A discharge of the "A" Low Level Waste Drain Tank (LLWDT) is commenced.
2. It is discovered that liquid waste radiation monitor 3LWS-RE70 is no longer functioning.
3. The crew terminates the discharge.
4. I&C is contacted, and they initiate best efforts to repair the instrument.

The crew desires to recommence discharging the "A" Low Level Waste Drain Tank

What additional actions are required in order to discharge the LLWDT with 3LWS-RE70 out of service?

a) Perform at least two independent samples, independent release calculations, and independent discharge valve lineups.
b) Direct I&C to install a temporary monitor with an alarm setpoint below that of the 3LWS-RE70 setpoint.
c) Recirculate the LLWDT an additional 15 minutes, and perform two independent discharge valve lineups.
d) Reconfirm release calculations. Direct Chemistry to take samples every 15 minutes while the discharge is in progress to ensure effluent is within limits.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since the applicant is required to have knowledge of administrative requirements concerning radiation hazards that may exist during normal or abnormal situations, and also apply REMODCM administrative requirements related to liquid radioactive release approvals.
"A" is correct, and "B", "C", and "D" wrong, since REMODCM Table V.C-1 ACTION A requires best efforts to repair the instrument (already in progress); and independent samples, release calculations, and discharge valve lineups prior to initiating a release.
"B" and "D" are plausible, since numerous actions with inoperable rad monitors or other discharge monitors involve temporary monitors or manual samples.
"C" is plausible, since recirculating the tank is required prior to its discharge.

Technical Reference(s): REMODCM (Rev. 30) Section V.C.1, page 122
(Attach if not previously provided, REMODCM (Rev. 30) Table V.C.-1 ACTION A, pg 123 and 124
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of LWS under high radiation condition detected by 3LWS-RE70 during a discharge.

Question Source: Bank #404705
Question History: Last NRC Exam Millstone 3 2019 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.11, 41.13, and 43.4
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 85 | Tier # | | 2 |
| K/A Statement: Reactor Protection System: Predict impact / mitigate failure of RPS signal to trip the reactor | Group # | | 1 |
| | K/A # | 012A2.06 | |
| Proposed Question: | Importance Rating | | 4.7 |

With the plant initially at 100% power, the following sequence of events occurs:

1. The "A" TDMFW Pump trips, and the MDMFW Pump will not start.
2. The SG Lo-Lo level setpoints are reached, but the Reactor does NOT trip.
3. The crew enters FR-S.1, *Response to Nuclear Power Generation/ATWS*.
4. Upon reaching FR-S.1, Step 18, "CHECK Reactor Subcritical", the following reports are made:
   - Reactor power: 60% and slowly decreasing.
   - All SG WR Levels: 30% and slowly decreasing.
   - CETs: 650°F and increasing.
5. Per FR-S.1, Step 18.RNO, the crew isolates steam release paths to allow the RCS to heat up.

What procedure action or transition is the US currently required to direct?

a) Go To E-0, *Reactor Trip or Safety Injection*.
b) Go To FR-H.1, *Response to Loss of Heat Sink*.
c) Go To SAG-01, *Main Control Room Severe Accident CR Guideline Initial Response*.
d) Remain in FR-S.1, *Response to Nuclear Power Generation/ATWS*. and return to step 4.

Proposed Answer: D

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, and apply detailed knowledge of FR-S.1 procedure flowpath and transition requirements that go beyond system knowledge and beyond knowing the overall strategy of the procedure.
"D" is correct, and "A", "B", and "C" wrong, since after isolating the steam release paths per the RNO in FR-S.1, step 18, and the reactor still at power, and CETCs <1200°F, the crew is directed to return to FR-S.1, step 4.
"A" is plausible, since all actions per FR-S.1 have been taken, and if power were below 5% with a negative SUR, FR-S.1, step 21 would direct the crew to "go to" the procedure and step in effect, which is E-0, and power is decreasing.
"B" is plausible, since after taking actions to allow the RCS to heat up, FR.S.1, step 18 RNO directs the crew to perform actions of other Function Restoration Procedures in effect which do not cool down or otherwise add positive reactivity to the core, while remaining in FR-S.1, and SG levels are well below entry conditions for FR-H.1.
"C" is plausible, since per FR-S.1, step 17, the crew would have been directed to "go to" SAG-01 if CETCs were $\geq$1200°F, and the crew has not been successful in shutting the reactor down in FR-S.1.

Technical Reference(s): FR-S.1 (Rev. 23), Steps 17-21
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss the conditions which require transition to other procedures from EOP 35 FR-S.1.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10 and 43.5
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 86 | Tier # | | 1 |
| K/A Statement: Loss of Residual Heat Removal System: | Group # | | 1 |
| Knowledge of annunciator alarms, indications, or response procedures. | K/A # | APE025G.2.4.31 | |
| | Importance Rating | | 4.1 |

Initial Conditions:

- RCS temperature is 180ºF.
- RCS pressure is 360 psia.
- There is a bubble in the Pressurizer.
- The "B" RCP is in service.
- The "A" RHR pump is in service.
- The "B" RHR pump is tagged out for breaker repairs.
- All S/Gs are at 45% NR level.

The following sequence of events occurs:

1. The RHR PUMP A FLOW LO annunciator is received on MB2.
2. The RO reports the "A" RHR Discharge Cold Leg Injection Valve (3SIL*MV8809A) has spuriously closed.
3. The crew is unable to open 3SIL*MV8809A.
4. The crew enters EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory*.

Per EOP 3505, what action is the US required to direct to restore Shutdown Cooling?

a) Maintain the “B” RCP running and establish forced circulation cooling by opening available SG Atmospheric Relief Valves.
b) Stop the “B" RCP and establish natural circulation cooling by opening available SG Atmospheric Relief Valves.
c) Open both PORVs and feed the RCS from the RWST using gravity feed via the RHR Pump Cold Leg Injection Valves.
d) Open both PORVs and feed the RCS from the RWST using one Charging Pump via the Charging Pump Cold Leg Injection Valves.

Proposed Answer: A

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, recognize OP 3505 applies, and determine the action to be taken per the correct EOP 3505 attachment. Per EOP 3505, Mode 5, non RIO conditions directs the operators to Att. "A" of 3505. "A" is correct, and “B”, “C”, and “D” are wrong, since Attachment A applies during Mode 5, non-RIO conditions, and Attachment A will establish forced circulation cooling using the running RCP.“B” is plausible, since if RCS pressure was less than 310 psia, operators would be required to stop the RCP and establish natural circulation cooling.“C” is plausible, since Gravity feed is utilized in Modes 6 or Zero (Att. "B"). “D” is plausible, since Feed & Bleed cooling is aligned if natural circulation attempts are unsuccessful.

Technical Reference(s): EOP 3505 (Rev. 17), Entry Conditions
(Attach if not previously provided, EOP 3505 (Rev. 17), Step 1-9
including version/revision number.) EOP 3505 (Rev. 17), Att. "A", steps 1-13

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per EOP 3505.

Question Source: Bank #407796

Question History: Last NRC Exam Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3 and 43.5

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 87 | Tier # | | 3 |
| K/A Statement: Knowledge of facility protection | Group # | | 4 |
| requirements, including fire brigade and portable fire fighting equipment usage | K/A # | G2.4.26 | |
| | Importance Rating | | 3.6 |

Proposed Question:

The plant is at 100% power on the nightshift, with the on-shift crew staffed as follows:

- One SM
- One US
- One STA
- Two ROs
- Five PEOs

The following sequence of events occurs:

1. A Security Guard calls the Control Room and reports a fire in the Cable Spreading Room.
2. The crew enters EOP 3509, *Fire Emergency*.
3. The Fire Brigade is being dispatched to fight the fire using the Fire Fighting Strategy Book.

Complete the following statement concerning how many of the five Millstone 3 PEO's are allowed to be dispatched as fire brigade members, and which action in the Fire Fighting Strategy Book specifically requires Shift Manager permission to perform.

____(1)___Millstone 3 PEOs are allowed to be dispatched as fire brigade members, and Shift Manager permission is required to____(2)___in the Cable Spreading Area.

| | (1) | (2) |
|---|---|---|
| a) | Three | pressurize the fire hoses |
| b) | Three | manually discharge the CO2 System |
| c) | Five | pressurize the fire hoses |
| d) | Five | manually discharge the CO2 System |

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level, since it requires applicants to have knowledge of Tech Spec and TRM Section 6 admin requirements on shift staffing, including the fire brigade. It also requires knowledge of SM responsibilities in the Fire Fighting Strategy Book, and Fire Brigade makeup per EOP 3509.
"C" and "D" are wrong, since two PEOs are required to meet minimum crew composition. This leaves three of the five PEOs available for the fire brigade. "C" and "D" are plausible, since five PEOs are required for the fire brigade, but this includes Millstone Unit 2 PEOs.
"B" is correct, since EOP 3509 will lockout the CO2 System for personnel safety, and the Fire Fighting Strategy Book requires Shift Manager permission to discharge CO2 into the area.
"A" is wrong, since Shift Manager permission is not required to pressurize the hoses. "A" is plausible, since the hose stations are kept dry and isolated in this area, due to concerns with all of the electrical cables in the area.

| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided, including version/revision number.) | Fire Fighting Strategies Book (Rev 00-0), Control Bldg, pg 32-35<br>Tech Spec (Amendment 212), Table 6.2-1<br>TRM (LBDCR 07-MP3-018) Requirement 6.2.2<br>EOP 3509 (Rev 26), step 2.g-i |

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04584 Describe the major administrative or procedural precautions and limitations placed on the operation of the CO2 Fire Protection (FPL) system…

| | | |
|---|---|---|
| Question Source: | Bank #403397 | |
| Question History: | Last NRC Exam | Millstone 3 2015 NRC Exam |
| Question Cognitive Level: | Memory or Fundamental Knowledge | |
| 10 CFR Part 55 Content: | 55.43.2 and 43.5 | |

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 88 | Tier # | | 1 |
| K/A Statement: Plant Fire On Site: Determine/interpret whether malfunction is due to common mode electrical failures | Group # | | 2 |
| | K/A # | APE067.AA2.07 | |
| Proposed Question: | Importance Rating | | 3.1 |

With the reactor initially at 100% power, the following sequence of events occurs:

1. A fire starts in the Instrument Rack Room.
2. The crew enters EOP 3509.1, *Control Room, Cable Spreading Area Or Instrument Rack Room Fire*.
3. The "A" MDAFW Pump spuriously starts.
4. The "A" Control Room Emergency Ventilation Supply Fan (3HVC*FN1A) spuriously starts.
5. The "B" MDAFW Pump spuriously starts.
6. The fire remains out of control.

Per EOP 3509.1, what action is required to be taken by the crew?

a) Trip the Reactor and transition to E-0, *Reactor Trip or Safety Injection* to respond to the Reactor trip while continuing to take actions in EOP 3509.1.
b) Continue taking actions in EOP 3509.1, since neither a Reactor trip nor a Control Room evacuation are currently required.
c) Remain in EOP 3509.1, and go to step 7 "Announce Control Room Evacuation", since a Control Room evacuation is required.
d) Remain in EOP 3509.1, and go to step 15 "Evacuate the Control Room", since an immediate Control Room evacuation is required.

Proposed Answer: C

Explanation (Optional):

This question is considered SRO level, since applicants are required to assess plant conditions, determine whether a shutdown is required, and determine whether to remain in the current section of the procedure in the Control Room, or transition to the portion of the procedure that directs the crew to evacuate the Control Room. This goes beyond system knowledge and beyond knowing the overall strategy of the procedure. "A" is wrong since step 8 note directs the operators not to use E-0 when they trip the reactor in preparation for a Control Room evacuation. "A" is plausible since Note prior to step 6 states that a transition to E-0 is allowed along with normal use of the EOPs, but this note is bypassed with spurious actuations on both trains.

"C" is correct, and "B" wrong, since EOP 3509.1 step 4 requires evacuation if BOTH trains of systems required for hot standby are affected. Since the multiple failures on equipment that is susceptible to the fire in this area, and both trains are affected, so the crew is required to proceed to the note prior to step 7 and evacuate the control room. "B" is plausible, since if the failures had been on equipment not related to achieving Hot Shutdown, or had been on only one train, this action would be correct.

"D" is wrong since proceeding to step 15 is only required if the evacuation is required immediately due to personnel safety concerns (hot gas, CO2, smoke, or flame). "D" is plausible since these actions would be required based on the severity of the fire, and the fire is currently out of control.

Technical Reference(s): EOP 3509.1 (Rev. 25), steps 1-15, especially steps 1, 3, 4, 7, and 15
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per EOP 3509

Question Source: Bank #407845
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.8 and 43.5
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 89 | Tier # | | 1 |
| K/A Statement: Ability to determine and interpret | Group # | | 2 |
| The following as it applies to AOP 3555, RCS | K/A # | Site Priority | 3555.A2.09 |
| Leak: RCS leak paths | Importance Rating | | 2.9 |

Proposed Question:

The plant is at 100% power, and current plant conditions are as follows:

- The crew has entered AOP 3555, *Reactor Coolant Leak.*
- The RO has throttled open 3CHS*FCV121 and stabilized pressurizer level on program.
- The STA estimates the RCS leak to be 35 gpm.

The following sequence of events occurs:

1. The US chooses to implement AOP 3555 step 13 “Isolate Charging and Letdown”, and directs the RO to perform Attachment F, Isolating Letdown While Supplying Seal Injection At Normal Operating Pressure.
2. The RO simultaneously closes 3CHS*FCV121 and Letdown Orifice Isolation Valve 3CHS*AV8149B, and then closes Charging Isolation Valve 3CHS*MV8106.
3. The RO reports that pressurizer level is again decreasing but at a slower rate than initial leak indications.

In accordance with AOP 3555, what action is the crew required to take?

a) CLOSE Charging Loop Isolation Valves 3CHS*AV8146 and 8147, Letdown Header Inner CTMT Isolation Valve 3CHS*CV8160, and Letdown Isolation Valve 3 RCS*LCV460, to attempt to isolate the leak.
b) Keep normal Charging and Letdown isolated, and verify the RPCCW System is intact by verifying RPCCW Surge Tank level is stable.
c) Establish normal charging and letdown by throttling 3CHS*FCV121 in MANUAL, and simultaneously OPENING Charging Isolation Valve 3CHS*MV8106 and Letdown Orifice Isolation Valve 3CHS*AV8149B.
d) Keep normal Charging and Letdown isolated, and establish Head Vent Letdown to the VCT using GA-14.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it involves an assessment of plant conditions and then selection of a procedure section to mitigate.
"A" is correct, since the leak is still active, as evidenced by pressurizer level decreasing at a slow rate. If the leak were isolated, PZR level would be increasing, since seal injection is still being supplied; and AOP 3555 directs expanding the isolation boundary of Charging and Letdown to continue attempts to isolate the leak. "B" and "D" are wrong, since Charging and Letdown would not be kept isolated unless the leak was isolated. "D" is plausible since these actions are specified in AOP 3555 if the leak was isolated. "B" is plausible, since RPCCW is the next leak isolation step in AOP 3555. "C" is wrong, but plausible, since charging and letdown would only be restored if the leak still exists after isolation steps specified in "A" were unsuccessful.

Technical Reference(s): AOP 3555 (Rev. 24), steps 1, 13, 14.
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None.

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per AOP 3555

Question Source: Bank# 406905

Question History: Last NRC Exam Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 90 | Tier # | | 1 |
| K/A Statement: Dropped Control Rod: Ability to use | Group # | | 2 |
| Plant Computers to evaluate system or component status | K/A # | APE003G.2.1.19 | |
| Proposed Question: | Importance Rating | | 3.8 |

With the plant initially at 100% power, a transient occurs, and the RO makes the following initial report at MB4:

- The ROD POSITION DEVIATION annunciator is lit on MB4.
- The ONE ROD BOTTOM annunciator is lit on MB4.
- Reactor power has promptly lowered, and is starting to recover.
- Tave has decreased and is stabilizing.
- DRPI indication for Rod H8 has been lost.
- No rod bottom LEDs are lit.

The following sequence of events occurs:

1. I&C reports the cause of the loss of DRPI indication for Rod H8 is a defective LED on MB4.
2. After one hour, affected rod H8 is declared INOPERABLE.

Complete the following statement.

The Rod Supervision program __(1)__ capable of verifying the specific position of Rod H8, and when completing the SHUTDOWN MARGIN Surveillance, __(2)__.

a) (1) IS
   (2) the maximum worth for a dropped rod listed on the Miscellaneous Core Data Sheet will be used for the worth of rod H8
b) (1) IS
   (2) no actual calculation will be required
c) (1) IS NOT
   (2) the maximum worth for a dropped rod listed on the Miscellaneous Core Data Sheet will be used for the worth of rod H8
d) (1) IS NOT
   (2) no actual calculation will be required

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to apply knowledge of Surveillance Requirements during abnormal plant conditions.
Tech Spec Surveillance Requirement 4.1.1.1.1.a requires a Shutdown Margin to be performed for an inoperable rod.
“A” and “C” are wrong, since no actual calculation will be required. This is because core design analysis shows that Shutdown Margin is acceptable with any single inoperable control rod and an assumed most reactive stuck rod. “A” and “C” are plausible, since a Shutdown Margin Surveillance is required to be completed, and an actual calculation would be required if two or more rods were declared inoperable. Also, and if a calculation was required, the worth of a dropped rod listed in the RE Curve and Data Book miscellaneous core data sheet.
“B” is correct, and “D” wrong, since the Rod Supervision Program does display individual rod height along with Group Demand position, and its input to individual rod height is from the DRPI coils (which supply both the DRPI LEDs on MB4 and Rod Supervision), and DRPI has not lost power.
“D” is plausible, since Rod Supervision also provides Group Demand information, which is from the step counter input, not DRPI, and a DRPI malfunction has occurred.

| | |
|---|---|
| Technical Reference(s): | OP 3209B (Rev. 11), Steps 2.3.2, 4.1.2, and 4.1.12 |
| (Attach if not previously provided, | Tech Spec Surveillance Requirement 4.1.1.1.1.a (Amend. 258) |
| including version/revision number.) | RE Curve and Data Book, RE-G-03 (MP3-21-00) |
| | OP 3670.1-004 (Rev. 00-02), Item 22 (Page 2) |

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Rod Position Indication System under the following Normal, Abnormal, and Emergency conditions… Stuck, Misaligned or Dropped Rod…

| | |
|---|---|
| Question Source: | New |
| Question History: | Last NRC Exam N/A |
| Question Cognitive Level: | Memory or Fundamental Knowledge |
| 10 CFR Part 55 Content: | 55.41.6 and 43.2 |

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 91 | Tier # | | 2 |
| K/A Statement: Containment Cooling System: | Group # | | 1 |
| Knowledge of EOP mitigation strategies. | K/A # | 022G2.4.6 | |
| Proposed Question: | Importance Rating | | 4.7 |

With the plant initially at 100% power when the following sequence of events occurs:

1. Offsite power is lost.
2. The BOP operator reports the "B" CRDM Fan has tripped.
3. The crew is currently at ES-0.2, *Natural Circulation Cooldown*, step 3, "Check Electrical Alignment".

Complete the following statement about how current conditions impact the upcoming natural circulation cooldown.

Per ES-0.2, the crew will be required to__________.

a) transition from step 3 of ES-0.2 to ES-0.3, *Natural Circulation Cooldown With Steam Void in Vessel (with RVLMS)*
b) complete the first 13 steps of ES-0.2, and then to ES-0.3, *Natural Circulation Cooldown With Steam Void in Vessel (with RVLMS)*
c) increase RCS subcooling to greater than 132°F, and then conduct the cooldown in the same manner as if two CRDM fans are running
d) align a Reactor Vessel Head Vent letdown path, and then conduct the cooldown in the same manner as if two CRDM fans are running

Proposed Answer: D

Explanation (Optional):
This question is considered a KA match since the CRDM cooling system cools the air exhausting from the CRDM cooling fans to cool the air as it returns to the Ctmt atmosphere, working with the Containment Air Recirculation System to maintain Ctmt temperature. This question is considered SRO level, since it requires the applicant to assess plant conditions during emergency conditions, determine whether a transition to a different emergency sub (ES) procedure is required, and determine the required action based on specific conditions that go beyond system knowledge or overall procedure mitigation strategies.
"A" and B" are wrong, since ES-0.3 is not entered unless a cooldown rate in excess of the rate that will create a void in the head is required. "A" and "B" are plausible, since there is a greater likelihood of drawing a void with only one CRDM fan running, and ES-0.3 is designed to accommodate void growth.
"C" is wrong, since 132°F subcooling is only a requirement if the head vent path cannot be aligned. "C" is plausible, since if a CRDM fan were not available, and the head vent path could not be aligned, a 132°F subcooling requirement would be required.
"D" is correct, based on ES-0.2 steps 4 and 5. CRDM cooling fans are run to provide additional cooling to the vessel head, which cools down slowly due to ambient losses while forced circulation is not available. The running CRDM fans prevent drawing a void in the vessel head during the cooldown. Without two CRDM cooling fans available, the head vent path is aligned to provide added head vent cooling which will prevent drawing a void in the head.

Technical Reference(s): ES-0.2 (Rev. 19-01), steps 3, 4, 5, Note prior to step 11, and 13
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per ES-0.2.

Question Source: Bank #408710

Question History: Last NRC Exam Millstone 3 2004 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 92 | Tier # | | 2 |
| K/A Statement: | Group # | | 1 |
| Instrument Air System: Knowledge of AOPs | K/A # | 078G2.4.11 | |
| Proposed Question: | Importance Rating | | 4.2 |

A plant Cooldown is in progress per OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- RCS temperature is 250°F.
- VCT level is stable at 48%.
- The "B" Train of RHR is in service in the "Cooldown" Mode.
- SG levels are being maintained with the MDAFW Pumps.

The following sequence of events occurs:

1. Instrument Air (IAS) pressure starts decreasing, and the crew enters AOP 3562, *Loss of Instrument Air*.
2. The RO reports IAS pressure has decreased to zero psig as indicated on 3IAS-PI29 (MB1).

Which of the below actions from AOP 3562 is the crew currently required to perform prior to the restoration of air pressure?

a) Using GA-14, establish Reactor Vessel Head Vent Letdown to the PRT.
b) Per AOP 3562, locally place control switches for Traveling Screens to AUTO.
c) Per AOP 3562, perform RCS makeup from the RWST.
d) Using OP 3322, stop the TDAFW Pump (3FWA*P2).

Proposed Answer: A

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions to determine which AOP and OP sections are currently applicable.
With IAS pressure at zero psig, the crew will progress through AOP 3562 until step 12, which checks IAS pressure either greater than 95 psig, or increasing. If not, the crew will return to step 3. This step divides the procedure into two portions. The first portion applies during the loss of air. The second portion (step 13 and beyond) applies after air pressure has been restored.
"A" is correct, since step 4.RNO directs the crew to use GA-14, Establish Reactor Vessel Head Vent Letdown to the PRT. This step is in the section of AOP 3562 that is currently in use, and the RNO is applicable based on current air pressure.
"B" is wrong, since step 18, which, directs the crew to locally place control switches for Traveling Screens to AUTO, is in the section of AOP 3562, is not currently in use. "B" is plausible, since this step is in AOP 3562, and screen dp indications are affected by a loss of air, so action with the screens is required.
"C" is wrong, since step 5.RNO only directs the crew to perform RCS makeup from the RWST to control VCT level if VCT level is below 40%. "C" is plausible, since this step is applicable in the current portion of AOP 3562.
"D" is wrong since the crew is not directed to stop the TDAFW pump until step 16, which is not applicable in the current section of the procedure.
"D" is plausible, since the TDAFW Pump will auto start on a loss of air, and it will be addressed in AOP 3562, step 3, but step 3 step throttles the TDAFW Pump flow control valves, and doesn't stop the pump.

Technical Reference(s): AOP 3562 (Rev. 17), steps 3, 4, 5, 12, 16, and 18
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per AOP 3562

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 93 | Tier # | | 1 |
| K/A Statement: Rediagnosis and SI Termination: | Group # | | 2 |
| Ability to explain and apply system limits and precautions | K/A # | WE01&02G.2.1.32 | |
| Proposed Question: | Importance Rating | | 4.0 |

The plant is in MODE 3, with a cooldown is in progress per OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- RCS Temperature is 360°F and lowering.
- The crew is preparing to enter MODE 4.
- The crew is preparing to place the "A" Train of RHR in service in the "Cooldown Mode" shortly after entering MODE 4.

The following sequence of events occurs:

1. A spurious automatic Safety Injection Actuation occurs.
2. The crew transitions from E-0, *Reactor Trip or Safety Injection* to ES-1.1, *SI Termination*.
3. Numerous Auxiliary Building radiation monitors go into alarm.
4. The US desires to enter ES-0.0, *Rediagnosis* due to the alarms that have just been received.

Complete the following statement about why the crew initially was not preparing to place both trains of RHR in service upon entry into MODE 4; and whether the crew is currently allowed to enter ES-0.0.

The crew was not initially allowed to place both trains of RHR in service in the Cooldown Mode in order to__(1)__; and the crew currently__(2)__allowed to enter ES-0.0.

| | (1) | (2) |
|---|---|---|
| a) | prevent exceeding RPCCW temperature limits | IS |
| b) | keep one train of RHR available for injection | IS |
| c) | prevent exceeding RPCCW temperature limits | IS NOT |
| d) | keep one train of RHR available for injection | IS NOT |

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level, since plant conditions must be assessed, EOP usage rules must be applied, and ES-0.0 entry options must be understood. The question is considered a KA match since it tests a RHR System limit that applies when preparing to align RHR in the Cooldown Mode.
"A" and "C" are wrong, since the concern which allows only one train of RHR to be aligned for plant cooldown when RCS temperature is greater than 260°F is that above 260°F, the RWST elevation head is not adequate to prevent flashing in the RHR suction piping. When a RHR train is aligned for cooldown, RHR suction temperature will rise to RCS temperature and pressure. If that train is subsequently realigned to the RWST for injection, suction pressure drops to RWST pressure, and flashing of hot water in the RHS pump and suction line may occur.
"A" and "C" are plausible, since there are numerous RPCCW temperature and flow limits that are a concern when RHR is aligned in the cooldown mode.
E-0 is applicable when the plant is in Modes 1, 2, and 3 when SI is not blocked and at least one Accumulator isolation Valve is open. Using E-0 in any other MODE requires a step by step evaluation to determine if a specified action is still applicable in the current plant conditions. Aux Building Radiation Alarms are not expected at this point in the recovery, and this is an entry condition to ECA-1.2, but the EOP steps that direct this transition are placed fairly late in E-0 and E-1, and not in ES-1.1. ES-0.0 does provide a transition to ECA-1.2.
"B" is correct, and "D" wrong, since the crew has exited E-0, and SIS has actuated. Per ES-0.0 Entry Conditions, ES-0.0 is applicable in MODEs 1-4 when RHR is not in service, so it is applicable in the current conditions. Per the EOP Users Guide, ES-0.0 has no symptoms or entry conditions and is entered solely based on operator judgment. Also, it can only be entered if SI is actuated and E-0 has been exited. All of these conditions are met.
"D" is plausible, since the plant is in an unusual situation, and several specific conditions must be met before entry into ES-0.0 is allowed.

| | |
|---|---|
| Technical Reference(s): | OP 3208 (Rev. 34), Caution prior to step 4.3.7 |
| (Attach if not previously provided, | E-0 (Rev. 34), Section 2 |
| including version/revision number.) | ES-0.0 (Rev. 07), Section B, and Step 3 RNO |
| | OP 3272 (Rev. 12), Section 3.2, page 9 |

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the conditions under which EOP ES-0.0 (Rediagnosis) can be used.

| | |
|---|---|
| Question Source: | New |
| Question History: | Last NRC Exam N/A |
| Question Cognitive Level: | Comprehension or Analysis |
| 10 CFR Part 55 Content: | 55.43.5 |
| Comments: | |

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 94 | Tier # | | 2 |
| K/A Statement: Service Water System: | Group # | | 1 |
| Predict impact / mitigate a loss of Service Water | K/A # | 076A2.01 | |
| Proposed Question: | Importance Rating | | 3.7 |

The plant is initially at 3% power with a plant startup in progress per OP 3203, *Plant Startup*, when the following sequence of events occurs:

1. The SERVICE WTR PP AUTO TRIP/OVERCURRENT annunciator is received on MB1.
2. The RO reports the following:
   - The "A" Service Water Pump has tripped.
   - The "C" Service Water Pump will NOT start.
   - The "B" and "D" Service Water Pumps are running.
3. The crew enters AOP 3560, *Loss of Service Water*.

Complete the following statement.

Per AOP 3560, step 2, "Check Status Of Service Water System", the crew will align the Service Water Supply Valves to TPCCW to obtain___(1)____, and then go to/remain in___(2)____to continue mitigating the event.

a) (1) 3SWP*MOV71A CLOSED and 3SWP*MOV71B OPEN
   (2) AOP 3561, *Loss of Reactor Plant Component Cooling Water*
b) (1) 3SWP*MOV71A CLOSED and 3SWP*MOV71B OPEN
   (2) AOP 3560, *Loss of Service Water*
c) (1) 3SWP*MOV71A and 3SWP*MOV71B both CLOSED
   (2) AOP 3561, *Loss of Reactor Plant Component Cooling Water*
d) (1) 3SWP*MOV71A and 3SWP*MOV71B both CLOSED
   (2) AOP 3560, *Loss of Service Water*

Proposed Answer: A

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions and transition out of the current AOP based on decision points in the AOP for Loss of Service Water.
On the loss of one train of Service Water, the crew is required to enter AOP 3560, and SWP low pressure may automatically start the second pump in the non-affected train, and automatically close TPCCW supply valves 3SWP*MOV71A and B. On the loss of a single train of SWP, AOP 3560, steps 2.a-f will attempt to start a SWP Pump in the affected train, and if unsuccessful, direct the crew to start the second SWP pump in the non-affected train. AOP then start a second SWP pump in the non-affected train (The RO reported it is already running).
"C" and "D" are wrong, since with two SWP Pumps running in the non-affected train, AOP 3560, steps 2.e-g will direct the crew to close the affected train TPCCW supply valve 3SWP*MOV71A to provide train separation for the operable train, and open the operating train TPCCW supply valve 3SWP*MOV71B to provide cooling to the TPCCW Heat Exchangers.
"C" and "D" are plausible, since if the second SWP pump in the operating train could not be started, AOP 3560, step 2.f.RNO would direct the crew to close both SWP supply valves MOV71A and B. Also, TPCCW is a non-safety related system, and per the initial conditions, the plant is in MODE 2 with the Main Turbine off-line.
"A" is correct and "B" wrong, since AOP 3560, step 2.h directs the crew to go to AOP 3561, *Loss of Reactor Plant Component Cooling Water*, since Service Water cooling has been lost to the affected train RPCCW Heat Exchanger.
"B" is plausible, since a Service Water Train has been lost, and the RPCCW System is functioning properly.

Technical Reference(s): AOP 3560 (Rev. 13), steps 2.a-h
(Attach if not previously provided, P&ID 133A (Rev. 44), B (Rev. 91), and D (Rev. 56)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per AOP 3560.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.5, and 43.5

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 95 | Tier # | | 3 |
| K/A Statement: Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, etc. | Group # | | 2 |
| | K/A # | GEN.2.2.18 | |
| | Importance Rating | | 3.9 |

Proposed Question:

The plant is in MODE 0, and current plant conditions are as follows:

- An "A" Train Electrical Outage is in progress.
- The "B" Spent Fuel Pool Cooling (SFC) Pump is in operation.
- An Electrician, carrying a beeper, is available to establish temporary power to the "A" SFC Pump.
- The "B" and "C" RPCCW Pumps are available.
- The "B" and "D" Service Water Pumps are available.

The "B" SFC Pump trips.

Complete the following statement:

The "A" SFC Pump ___(1)___ allowed to be credited when determining Shutdown Risk (SDR) Condition Colors; and ___(2)___ considered available for use in EOP 3505A, *Loss of Spent Fuel Pool Cooling*.

| | (1) | (2) |
|---|---|---|
| a) | IS | IS |
| b) | IS | IS NOT |
| c) | IS NOT | IS |
| d) | IS NOT | IS NOT |

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions and determine whether Fuel Pool Cooling Pump administrative requirements are adequate to meet both Shutdown Risk administrative requirements during refueling and also whether it complies with availability requirements for a 3500 series EOP. Per Tech Spec Basis 3/4.9.1, MODE 0 is defined as "the Operational MODE where all fuel assemblies have been removed from containment to the Spent Fuel Pool." The "B" SFC Pump is available for EOP 3505A, since it does have its administrative requirements (electrician, 2 CCP pumps, and 2 SWP pumps) met ("B" and "D" wrong), but it is not credited for SDR, since it does not have its normal power supply available ("A" wrong, "C" correct). "A" is plausible, since the "A" SFC Pump does have its admin requirements met to use backup power. "B" and "D" are plausible, since normal power is not available, and there are several admin requirements needed to credit the pump.

Technical Reference(s): OU-M3-201 (Rev. 27), Att. 2, Page 5
(Attach if not previously provided, EOP 3505A (Rev. 15), Att. E
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the key safety functions monitored in OU-M3-201, Shutdown Safety Assessment Checklist

Question Source: Bank #407825

Question History: Last NRC Exam Millstone 3 2013 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 41.10, 43.5, 45.13

Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 96 | Tier # | | 1 |
| K/A Statement: Reactor Coolant Pump Malfunctions: | Group # | | 1 |
| Ability to perform specific system and integrated plant | K/A # | APE015G.2.1.23 | |
| procedures during all modes of plant operation. | Importance Rating | | 4.4 |

Proposed Question:

The plant initially at 35% power with an up-power in progress per OP 3204, *At Power Operations*, when the following sequence of events occurs:

1. The RCP HI RANGE LKG FLOW HI annunciator is received on MB3B, 2-10.
2. The RO reports "A" RCP # 1 seal return (CBO) flow has increased from 2.7 gpm to 4.5 gpm over the past 3 minutes.
3. The RCP A MID STG INLET PRESS HI Annunciator is received on MB4.
4. The RO reports "A" RCP Mid Stage Inlet Pressure is 1880 psig and stable.
5. The STA reports all other parameters and annunciators associated with the "A" RCP are normal.

What specific action is the crew directed to take per OP 3353.MB3B, 2-10?

a) Trip the Reactor, stop the "A" RCP, and go to E-0, *Reactor Trip or Safety Injection*.
b) Enter AOP 3554, *RCP Trip or Stopping a RCP at Power*, remove the RCP from service and isolate the number 1 seal within 5 minutes.
c) Commence an orderly plant shutdown and when in MODE 3, remove the "A" RCP from service.
d) Notify the OMOC (Duty Officer) and request Engineering Department evaluate continued pump operation.

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, and determine to move forward in the current procedure rather than transition out of it, and select the appropriate action based on abnormal conditions that go beyond system knowledge.
The normal seal return flow is about 2.7 gpm. The annunciator comes in at >3 gpm.
"C" is correct, and "A", "B", and "D" wrong, since with $\geq$4 gpm CBO flow, AND either the Mid OR Upper Stage Inlet Press Hi annunciators lit, the crew is required to perform an orderly plant shutdown and when in MODE 3, remove the RCP from service.
"A", "B", and "D" are plausible, since these actions could be directed per the ARP based on different RCP conditions.

Technical Reference(s): (Attach if not previously provided, including version/revision number.) OP 3353.MB3B, 2-10, steps 4, 12, and 15

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a plant condition requiring the use of the OP-3201 procedure, identify applicable technical specification action requirements.

Question Source: Bank #406864
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3 and 43.5.
Comments:

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 97 | Tier # | | 1 |
| K/A Statement: Determine / interpret the existence of a SG tube rupture, and its potential consequences | Group # | | 1 |
| | K/A # | EPE038EA2.02 | |
| Proposed Question: | Importance Rating | | 4.8 |

Safety Injection actuates due to Low Pressurizer Pressure, and the following sequence of events occurs:

| Time (Minutes) | Action |
|---|---|
| 0 | The crew enters E-0, *Reactor Trip or Safety Injection*. |
| 12 | The BOP reports the "A" SG level is increasing in an uncontrolled manner. |
| 20 | The BOP operator isolates AFW flow to the "A" SG with NR level at 28%. |
| 21 | The "A" SG Atmospheric Relief Valve fails open. |
| 43 | The BOP notices the failed open Atmospheric Relief Valve, and closes the valve. |
| 45 | Safety Injection is terminated. |

Complete the following statement concerning "operator credited actions" assumed by the FSAR for this event.

The operators were too slow to___(1)___, and the potential adverse consequence due to this delay is___(2)___.

a) (1) isolate AFW Flow to the "A" Steam Generator
   (2) exceeding the 10CFR fuel clad temperature limit
b) (1) isolate AFW Flow to the "A" Steam Generator
   (2) rad release to the environment exceeding off-site dose estimates
c) (1) close the "A" SG Atmospheric Relief Valve
   (2) exceeding RCS boundary thermal stress limits
d) (1) close the "A" SG Atmospheric Relief Valve
   (2) rad release to the environment exceeding off-site dose estimates

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires knowledge of the FSAR accident analysis beyond system knowledge. The Millstone 3 license (10CFR55.43.1) requires compliance with the FSAR and with Tech Specs. Tech Specs are included in 10CFR55.43.2, and compliance with the requirements of the FSAR can be included with 10CFR55.43.1. The License states that Millstone… Unit No. 3... is described in the licensees' "Final Safety Analysis Report." The FSAR assumes operators comply with certain action times to ensure the accident analysis is within acceptable limits, and the US is responsible for driving through the EOPs at a rate that will comply with these times. This event can be diagnosed as a SGTR based on low Pzr pressure and SG level increasing in an uncontrolled manner. Two operator credited actions that are required to be met to comply with the FSAR for the SGTR event are: a failed open atmospheric relief valve needs to be closed by the operators within 20 minutes (to remain within rad release assumptions), and isolating AFW flow to the ruptured SG by 30% Narrow range level, to comply with SG overfill analysis. Operators met the SG level requirement for isolating AFW ("A" and "B" wrong), but did not close the atmospheric relief valve within the 20 minute requirement. "A" and "B" are plausible, since the crew delayed in isolating AFW to the ruptured SG, which is allowed to be isolated when SG NR level exceeded 7%. The FSAR indicates that a significant partitioning factor that exists between SG water and SG steam is used in calculating off-site dose during the design basis SGTR.
"D" is correct, and "C" wrong, since the basis for closing the failed open atmospheric relief valve and for isolating AFW is to minimize radiation release, and ECCS flow and RWST inventory will keep the core covered during a SGTR. "A" is plausible, since a SGTR is in progress, RCS inventory is decreasing, and 2200°F clad temperature is a 10CFR design criterion on a loss of coolant.

| | |
|---|---|
| Technical Reference(s): (Attach if not previously provided, including version/revision number) | FSAR Chapter 15.6.3 (Rev. 30), Section 15.6.3.1<br>FSAR Chapter 15.6.3 (Rev. 30), Section 15.6.3.2.1<br>FSAR Chapter 15.6.3.2.2 (Rev. 30), Radiological Consequences<br>FSAR Table 15.6.3-1 (Rev. 30) Operator Action Times |

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04921 Outline the anticipated Operator Actions in response to SGTRs to include the operator credited actions in FSAR chapter 15. (As available)

Question Source: Modified Bank #410186

Question History: N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.1 and 43.5

Comments:

This question is considered “modified” since the time the atmospheric relief valve was open has been changed, as well as the time it took to terminate SIS, making a new correct answer.

Original Bank Question

Safety Injection actuates due to Low Pressurizer Pressure, and the following sequence of events occurs:

| Time (Minutes) | Action |
|---|---|
| 0 | The crew enters E-0, *Reactor Trip or Safety Injection*. |
| 12 | The BOP reports the “A” SG level is increasing in an uncontrolled manner. |
| 25 | The BOP operator isolates AFW flow to the “A” SG with NR level at 31%. |
| 27 | The “A” SG Atmospheric Relief Valve fails open. |
| 40 | The BOP notices the failed open Atmospheric Relief Valve, and closes the valve. |
| 45 | Safety Injection is terminated. |

Complete the following statement concerning "operator credited actions" assumed by the FSAR for this event.

The operators were too slow to __(1)___, and the potential adverse consequence due to this delay is__(2)___.

a) (1) isolate AFW Flow to the “A” Steam Generator
   (2) exceeding the 10CFR fuel clad temperature limit
b) (1) isolate AFW Flow to the “A” Steam Generator
   (2) rad release to the environment exceeding off-site dose estimates
c) (1) close the “A” SG Atmospheric Relief Valve
   (2) exceeding RCS boundary thermal stress limits
d) (1) close the “A” SG Atmospheric Relief Valve
   (2) rad release to the environment exceeding off-site dose estimates

Answer: B

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 98 | Tier # | | 3 |
| K/A Statement: | Group # | | 2 |
| Knowledge of the process for making changes to procedures. | K/A # | GEN.2.2.6 | |
| | Importance Rating | | 3.6 |

Proposed Question:

An on-shift SRO is reviewing a draft Surveillance Procedure change.

The change consists of modifying an initial condition.

Complete the following statement per AD-AA-100, *Technical Procedure Process Control*.

This change is classified as a/an__(1)___Change; and the SRO___(2)___authorized to sign for Cognizant Management approval.

| | (1) | (2) |
|---|---|---|
| a) | NON-INTENT | IS |
| b) | NON-INTENT | IS NOT |
| c) | INTENT | IS |
| d) | INTENT | IS NOT |

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires knowledge of administrative procedures that facilitate proper implementation of normal plant procedures.
This procedure revision changes an initial condition and therefore is required to be classified as an "INTENT" change (per 5.3.16 of AD-AA-100). Additionally, AD-AA-100 Attachment 19, requires Cognizant Management "B" approval for "INTENT" changes. For Operation's procedures, this is either the Manager of Nuclear Operations or OMOC ("D" correct / "A", "B' & "C" wrong).
"A" is plausible as a "NON-INTENT" change allows for approval by an Operations Department Active SRO.
"B" is plausible as many changes are approved the Manager of Nuclear Operations or OMOC.
"C" is plausible as SRO's have approval authority in the procedure change process.

Technical Reference(s): AD-AA-100 (Rev. 13), step 5.3.16 (pg 82)
(Attach if not previously provided, AD-AA-100 (Rev. 13), Attachment 1 (pg 93)
including version/revision number.) AD-AA-100 (Rev. 13), Attachment 19 (pg 119)

Proposed references to be provided to applicants during examination: None.

Learning Objective: Outline the process for modifying a document

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.10, 43.3, 45.13

Comments:

| Examination Outline Cross-reference: | Level | RO | SRO |
|---|---|---|---|
| Question # 99 | Tier # | | 3 |
| K/A Statement: Ability to use procedures to determine the effects on reactivity of plant changes such as reactor coolant system temperature, secondary plant, fuel depletion, etc. | Group # | | 1 |
| | K/A # | G2.1.43 | |
| | Importance Rating | | 4.3 |

Proposed Question:

The crew is preparing to perform a Xenon-free reactor startup per OP 3202, *Reactor Startup*, and initial conditions are as follows:

- Burnup is 15,000 MWD/MTU.
- The selected ECC rod height is Control Bank D at 90 steps.

Prior to commencing the withdrawal of rods, an unobserved dilution event occurs, reducing RCS boron concentration by 200 ppm.

**Using the attached Rod Worth and Boron Worth curves,** complete the statements below, assuming the ECC was accurate prior to the dilution event occurring.

As criticality is approached, the 1/M plot will predict criticality to occur__(1)__RIL; and based on the effects of this unobserved dilution, per OP 3202, the crew will be required to__(2)__.

a) (1) above
   (2) continue with the startup, and initiate a CR to track the reactivity management event
b) (1) above
   (2) insert all control banks back into the core and recalculate the ECC
c) (1) below
   (2) commence immediate boration per AOP 3566, *Immediate Boration*, and insert rods
d) (1) below
   (2) trip the Reactor and enter E-0, *Reactor Trip or Safety Injection*

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to have knowledge of administrative procedures involved in specifying actions based on changes in core reactivity, as well as knowledge of actions based on Tech Spec limits.
Calculation: 15,000 MWD/MTU is MOL. Per Curve RE-D-03, IRW at ECP (CBD @ 90 steps) is 533 pcm. IRW at the 500 pcm admin limit is 533 pcm + 500 pcm = 1033 pcm. Per Curve RE-D-03, IRW at RIL (CBC @ 51 steps) is 1802 pcm. The reactivity difference between RIL and ECC is 1802 pcm – 533 pcm = 1269 pcm. Per Curve RE-F-02, Zero Power DBW is -7.05 pcm/ppm. The dilution required to drop from the ECC to RIL is 1269 pcm / -7.05 pcm/ppm = 180 ppm.
"A" and "B" are wrong, since the change in boron concentration is greater than 180 ppm, so the prediction will show the reactor going critical below RIL. "A" and "B" are plausible, since the admin limit is also exceeded, and the listed actions would be required if RIL was not also exceeded.
"C" is correct, and "D" wrong, since OP 3203 requires the operators to perform an immediate boration and fully insert the control rods back into the core. "D" is plausible, since criticality below RIL is a violation of Tech Specs, and OP 3203 does have reactor trip criteria.

| | |
|---|---|
| Technical Reference(s): | OP 3202 (Rev.24), Steps 3,14, 4.27, and 4.30 |
| (Attach if not previously provided, | OP 3209A (Rev.10), step 4.2.5 |
| including version/revision number.) | U3 TRM (LBDCR 17-MP3-007), Figure 6 on page 8.1-16 |
| | RE Curve and Data Book, Curve RE-D-03, MOL (MP3-21-00) |
| | RE Curve and Data Book, Curve RE-F-02 (MP3-21-00) |

Proposed references to be provided to applicants during examination: **RE-D-03 MOL Curves**
**RE-F-02 Curves**

Learning Objective: State the minimum margin between the ECC and RIL, and the basis for that margin

| | |
|---|---|
| Question Source: | Modified Bank #406541 |
| Question History: | Last NRC Exam N/A |
| Question Cognitive Level: | Comprehension or Analysis |
| 10 CFR Part 55 Content: | 55.3.5 and 43.6 |

Comments:

This question is considered “modified” since the amount of the dilution has changed significantly, changing an old distractor to the new correct answer. Also, all curves have changed, changing all of the numbers in the calculation.

Original bank question:

The crew is preparing to perform a Xenon-free reactor startup per OP 3202, *Reactor Startup*, and initial conditions are as follows:

- Burnup is 15,000 MWD/MTU.
- The selected ECC rod height is Control Bank D at 90 steps.

Prior to commencing the withdrawal of rods, an unobserved dilution event occurs, reducing RCS boron concentration by 100 ppm.

**Using the attached Rod Worth and Boron Worth curves,** complete the statements below, assuming the ECC was accurate prior to the dilution event occurring.

As criticality is approached, the 1/M plot will predict criticality to occur __(1)__ RIL; and based on the effects of this unobserved dilution, per OP 3202, the crew will be required/allowed to __(2)__.

a) (1) below
(2) trip the Reactor and enter E-0, *Reactor Trip or Safety Injection*
b) (1) below
(2) commence immediate boration per AOP 3566, *Immediate Boration*
c) (1) above
(2) insert all control banks back into the core and recalculate the ECC
d) (1) above
(2) continue the startup, and initiate a CR to track the reactivity management event

Answer: C

| | | | |
|---|---|---|---|
| Examination Outline Cross-reference: | Level | RO | SRO |
| Question # 100 | Tier # | | 1 |
| K/A Statement: Station Blackout: Determine / interpret faults and lockouts that must be cleared prior to re-energizing buses | Group # | | 1 |
| | K/A # | EPE055EA2.06 | |
| Proposed Question: | Importance Rating | | 4.1 |

With the plant initially at 100% power, offsite power becomes unstable, resulting in the following sequence of events occurs:

1. A "NSST Primary Lockout" Annunciator illuminates on MB8.
2. Offsite power is lost.
3. The crew enters ECA-0.0*, Loss of All AC Power*.
4. The crew NOT successful at restoring power via either the EDGs or the SBO Diesel.
5. The crew commences depressurizing all four SGs to cooldown and depressurize the RCS.
6. Offsite power is restored, and the crew is preparing to reenergize 4KV Bus 34C from offsite power.

Complete the following statement:

The crew__(1)__need to clear the NSST Primary Lockout to allow them to re-energize Bus 34C; and when power is restored to Bus 34C, the first action the US will direct the crew to take is to__(2)__.

| | (1) | (2) |
|---|---|---|
| a) | DOES | move ahead in ECA-0.0, and "Stabilize SG pressures". |
| b) | DOES | transition to ECA-0.1, *Loss of All AC Power – Recovery Without SI Required* |
| c) | DOES NOT | move ahead in ECA-0.0, and "Stabilize SG pressures" |
| d) | DOES NOT | transition to ECA-0.1, *Loss of All AC Power – Recovery Without SI Required* |

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, and determine to move forward in the current procedure rather than transition out of it, and select the appropriate action based on abnormal conditions that go beyond system knowledge and beyond knowing the overall strategy of this Emergency Contingency Action (ECA) procedure.
"A" and "B" are wrong, since a NSST Primary Lockout does not prevent the closing the RSST on to 4KV Bus 34C, and ECA-0.0 allows restoration of offsite power from either the NSSTs or RSSTs. "A" and "B" are plausible, since a NSST Primary Lockout trips the Main Turbine, and trips and locks out numerous breakers, including the Main Generator Output Breaker, the Generator Field Breaker, and the NSST feeder breakers to 4KV buses 34A and B, and NSST feeder breakers to 6.9 KV Buses 35A, B, C, and D.
"C" is correct, and "D" wrong, since when power is restored to an emergency bus from offsite power per step 8 (continuous action step) after SG depressurization has commenced (step 20), the crew will move forward in ECA-0.0 to step 28, and if power has been restored to an emergency bus, the crew will proceed to step 29 and stabilize SG pressures prior to transitioning to ECA-0.1 or 0.2 in step 33. "D" is plausible, since the crew will be transitioning to ECA-0.1 shortly after stabilizing SG pressures to restore plant equipment in the proper priority to regain plant control after power is restored.

| | |
|---|---|
| Technical Reference(s): | ECA-0.0 (Rev. 40), Steps 8.a.RNO, 20, 28, 29, and 33 |
| (Attach if not previously provided, | OP 3353.MB8A (Rev. 11), 2-6, Automatic Functions |
| including version/revision number.) | EE-1A (Rev. 27) |
| | LSK-24-3D (Rev. 12) |

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per EOP 35 ECA-0.0.

| | | |
|---|---|---|
| Question Source: | New | |
| Question History: | Last NRC Exam | N/A |
| Question Cognitive Level: | Comprehension or Analysis | |
| 10 CFR Part 55 Content: | 55.41.6 and 43.5 | |

Comments: