Examination Outline Cross-reference:	Level	RO	SRO
Question # 1	Tier #	1	1
Reactor Trip, Stabilization, Recovery: Ability to	Group #	1	1
determine and/or interpret lights and alarms	K/A #	EPE 7.EA2.05	
Proposed Question:	Importance Rating	3.7	

With the plant initially at 100% power, the Reactor trips 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.
b) The DWST LEVEL LO annunciator should not be lit, since the alarm setpoint indicates the DWST is about 300,000 gallons, which is below its Tech Spec minimum water level.
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.
d) The MOIST SEP LEVEL HI annunciators should not be lit, since the "A" Moisture Separator Drain Tank Emergency Drain Valve should have opened.

Proposed Answer: D

Explanation: "D" is correct, since high level in the MSR Drain Tank sends a signal to open the emergency drain valve, preventing a Moisture Separator high level condition. 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 the AFW suctions are normally aligned to the DWST at power, and they auto-realign to the DWST on SG Lo-Lo level, which occurs on a trip from high power levels. The DWST is a 360,000 gallon tank, its Tech Spec minimum level in MODES 1-3 is 334,000 gallons, and the DWST LEVEL LO annunciator setpoint is 336,500 gallons. Total AFW flow on a trip is 1,200 gallons per minute. Since the DWST Tech Spec minimum level is below the DWST LEVEL LO annunciator setpoint, this annunciator could be received shortly after the trip. And the longest time it could take to reach the DWST LEVEL LO alarm without operator action is (360,000 gal – 336,500 gal) ÷ 1,200 gal / minute = 19.6 minutes. "B" is plausible, since the DWST LEVEL LO-LO annunciator, which has an alarm setpoint of 31,800 gallons, should not be reached for several hours without operator action.
"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):	OP3353.MB5A, 5-5 (Rev. 15), and MB5A, 5-5 (Rev. 15)
(Attach if not previously provided,	OP3353.MB5C, 5-3 (Rev. 16), and MB6A, 1-1B (Rev. 24)
including version/revision number.)	OP3353.MB6A, 2-1 (Rev. 24), and MB6A, 3-4 (Rev. 24)
	OP 3353.MB6A, 5-8 (Rev. 24), and MB6B, 2-8A (Rev. 00)
	OP 3322 (Rev. 37), Step 4.1.2.d
	Tech Spec LCO 3.7.1.3 (Amendment 258)

P&ID 125A (Rev. 35)

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 #403912
Question History: Last NRC Exam Millstone 3 2021 NRC Exam
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 # 2	Tier #	1	1
K/A Statement: Large Break LOCA: Knowledge of	Group #	1	1
the reasons for actions contained in an EOP	K/A #	EPE 11.EK3.12	
Proposed Question:	Importance Rating	4.2	

A large break LOCA occurs, and initial conditions are as follows:

1. The crew has just entered ES-1.3, *Transfer to Cold Leg Recirculation*.
2. The "A" RHR Pump is still running.

Per ES-1.3, step 2, the crew stops the "A" RHR Pump.

Complete the following statement.

The bases for tripping the running RHR pump(s) in ES-1.3, step 1 is to reserve the remaining RWST water for____________________________.

a) maintaining adequate suction to the CHS and SIH pumps while switching to cold leg recirc, and for Quench Spray Pump usage after the switchover is complete.
b) preventing cavitation of the RHR pumps, and for Quench Spray Pump usage after the switchover is complete.
c) maintaining adequate suction to the CHS and SIH pumps while switching to cold leg recirc, and for minimizing boron precipitation in the hottest regions of the core.
d) preventing cavitation of the RHR pumps, and for minimizing boron precipitation in the hottest regions of the core.

Proposed Answer: A

Explanation:
"A" is correct since, when RWST level decreases to the switchover setpoint, the transfer to cold leg recirculation is made to maintain coolant flow to the core (This is a time-critical operator action, to ensure the switchover is complete prior to losing suction on the CHS/SIH pumps, and the remainder of the RWST is reserved for QSS pump usage for lowering CTMT pressure.
"B" and "D" are wrong, since the RHR pumps trip are no longer needed for decay heat removal by the time the RWST Lo-Lo level setpoint is reached. "B" and "D" are plausible since RHR pumps initially provide core cooling by taking suction on the RWST.
"C" and "D" are wrong, since boron precipitation is not a concern for the first several hours after the onset of a large break LOCA. "C" and "D" are plausible, since boron precipitation is the basis for switching to hot leg recirc.

Technical Reference(s): WOG Bkgd Doc for ES-1.2 (Rev. 2), Caution prior to step 1
(Attach if not previously provided, Millstone 3 ES-1.3 Basis Document (Rev. 20), step 2
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 ES-1.3
Question Source: Bank #407936
Question History: Last NRC Exam Millstone 3 2007 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5, 41.8, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 3	Tier #	1	1
K/A Statement: Reactor Coolant Pump Malfunctions: Ability to determine and/or interpret RCP high stator temperature	Group #	1	1
	K/A #	APE 15.AA2.09	
	Importance Rating	3.3	

Proposed Question:

Initial conditions:

- A plant shutdown is in progress per OP 3206, *Plant Shutdown*.
- The Reactor is at 6% power.

The following sequence of events occurs:

1. The RCP D MOTOR TEMP HI Annunciator is received on MB4B, window 2-7.
2. The RO reports the computer point indicates the "D" RCP Stator temperature is 320°F.

Per OP 3353.MB4B, 2-7, what action is required to be taken by the crew?

a) Verify Chilled Water is aligned to the "D" RCP motor air cooler, and continue the plant shutdown per OP 3206, *Plant Shutdown*.
b) Stop the "D" RCP, and continue the plant shutdown per OP 3206, *Plant Shutdown*.
c) Go to AOP 3554, *RCP Trip or Stopping an RCP at Power*, and remove the "D" RCP from service per AOP 3554.
d) Trip the Reactor, stop the "D" RCP, and go to E-0, *Reactor Trip or Safety Injection*.

Proposed Answer: C

Explanation:
The ARP requires the crew to confirm the alarm (current temperature is above the alarm setpoint of 302°F).
"C" is correct, and "A", "B", and "D" wrong, since with the high temperature alarm confirmed, temperature greater than 311°F, and power below P-10 (11%), the ARP directs the crew to go to AOP 3554 and remove the RCP from service.
"A" is plausible, since the ARP directs the crew to confirm the RCP high temperature prior to stopping the RCP, and Chilled Water normally supplies the RCP motor air coolers, but this cooler is on the exhaust side of the cooling air to the motor.
"B" is plausible, since a shutdown is in progress, and the RCP can be removed from service without causing a Reactor trip.
"D" is plausible, since this action would be required by the ARP if power were above P-10 (11%).

Technical Reference(s): OP 3353.MB4B, 2-7 (Rev. 11), Steps 1, 2.1, and 2.2
(Attach if not previously provided, AOP 3554 (Rev. 11), Section 2.2, and Steps 1-3
including version/revision number.) Tech Spec Table 2.2-1 (Amendment 242), Functional Unit 18
Proposed references to be provided to applicants during examination: None
Learning Objective: Identify plant conditions that require entry into AOP 3554
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 # 4	Tier #	1	1
K/A Statement: Loss of Reactor Coolant Makeup:	Group #	1	1
Ability to determine and/or interpret whether a Charging line leak exists	K/A #	APE 22.AA2.01	
	Importance Rating	3.3	

Proposed Question:

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

<u>Time (minutes)</u>	<u>Event</u>
T=0:	Pressurizer level starts slowly decreasing.
T=0:	The RO reports indicated Charging flow has decreased below its original value of 55 gpm.
T+3:	The crew enters AOP 3555, *Reactor Coolant Leak*.
T+5:	The crew stabilizes the plant per AOP 3555.
T+6:	The RO reports <u>indicated</u> Charging flow at MB3 is back at its original value of 55 gpm.
T+7:	The STA performs a mass balance, and estimates the leak is 10 GPM.
T+10:	The crew is preparing to isolate Charging and Letdown while looking for the source of the leak.

Complete the following statement, assuming the plant is still at 100% power, the leak is still active, and no further operator actions will be taken.

The leak is located______(1)______ of Charging Line Flow Transmitter 3CHS-FT121, and indicated______(2)______at MB3/4.

	(1)	(2)
a)	upstream	VCT level is decreasing (between makeups)
b)	downstream	VCT level is decreasing (between makeups)
c)	upstream	Pressurizer level is decreasing
d)	downstream	Pressurizer level is decreasing

Proposed Answer: A

Explanation:
An RCS leak initially results in Pressurizer level decreasing, since mass out of the RCS now exceeds mass in from Charging. Per AOP 3555, the crew will manually throttle open Charging Line Flow Control Valve 3CHS*FCV121 to stabilize Pzr level and then maintain level on program. This will require increased Charging flow to compensate for the leak.
"B" is correct, since with indicated charging flow back at its original value and the leak still exists. The flow transmitter is not sensing the increased Charging flow required to maintain Pzr level and compensate for water being lost out the break. Also, the increase in charging flow required to maintain Pressurizer level stable will lower VCT level. A 10-gpm leak is within the capacity of the VCT level control system, so VCT level will decrease between makeups, and remain within its normal band with increased makeup frequency.
"B" is wrong, since excess charging flow is required to maintain Pzr level stable and compensate for the inventory being lost out the break, and this excess flow is not being sensed by FT121. "B" is plausible, since Pzr level is stable.
"C" and "D" are wrong, since per AOP 3555, the crew has stabilized Pzr level by increasing Charging flow. "C" and "D" are plausible, since initially, Pzr level was decreasing.

Technical Reference(s): AOP 3555 (Rev. 25), Steps 1-3
(Attach if not previously provided, P&ID 104A (Rev. 58)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Primary Makeup System under Normal Operations, Abnormal Operations, and Emergency Operations.

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 41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 5	Tier #	1	1
K/A Statement: Loss of RHR System:	Group #	1	1
Ability to operate and/or monitor ECCS	K/A #	APE 25.AA1.20	
Proposed Question:	Importance Rating	3.7	

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

- The Pressurizer is solid.
- COPPS is ARMED.
- No RCPs are running.
- Train "B" electrical outage is in progress.

The following sequence of events occurs:

1. The "A" RHR Pump trips and cannot be restarted.
2. The crew enters EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventor*y.
3. Per EOP 3505 Attachment A, the crew is preparing to initiate RCS Bleed and Feed cooling.

Complete the following statement:

The RO will align the bleed path by ___(1)____, and align the preferred feed path from the RWST by using the "A"____(2)______.

a) (1) manually opening both Pzr PORVs
   (2) Charging Pump
b) (1) allowing RCS pressure to increase to the PORV COPPS setpoint
   (2) Charging Pump
c) (1) manually opening both Pzr PORVs
   (2) Safety Injection Pump
d) (1) allowing RCS pressure to increase to the PORV COPPS setpoint
   (2) Safety Injection Pump

Proposed Answer: A

Explanation: This question is considered a KA match, since at Millstone 3, the Charging Pumps serve as the High Head Safety Injection Pumps.
"C" and "D" are wrong, since a Charging is the preferred pump for feeding the RCS from the RWST.
"C" and "D" are plausible, since if a Charging Pump is not available, a SIH pump will be used.
"A" is correct, and "B" wrong, since the RO will be directed to manually open both PORVs to align the bleed path. "B" is plausible, since the Pressurizer is solid, and a Charging Pump will quickly raise RCS pressure to the PORV setpoint, which is significantly lowered when COPPS is Armed.

Technical Reference(s): EOP 3505 (Rev 17), Att. A, step A.11, including Note
(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, properly apply the notes and cautions of EOP 3505

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, 41.5. 41.7 41.8, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 6	Tier #	1	1
K/A Statement: Loss of Component Cooling Water:	Group #	1	1
Operational implications and/or cause and effect	K/A #	APE 26.AK1.02	
relationships of a loss of cooling to CCWS	Importance Rating	3.8	

Proposed Question:

The plant is at 100% power.

Service Water to “A” RPCCW Heat Exchanger Supply Valve 3SWP*MOV50A spuriously closes.

Which Heat Exchanger has lost cooling?

a) The Excess Letdown Heat Exchanger
b) The "C" RCP Thermal Barrier Heat Exchanger
c) The Letdown Heat Exchanger
d) The Seal Water Heat Exchanger

Proposed Answer: C

Explanation:
"A" is wrong, since the Excess Letdown Heat Exchanger is cooled by Train "B" RPCCW.
“B” is wrong, since the "C" RCP Thermal Barrier Heat Exchanger is cooled by Train "B" RPCCW.
"C" is correct, since the Letdown Heat Exchanger is cooled by Train "A" RPCCW.
"D" is wrong, since the Seal Water Heat Exchanger is cooled by Train "B" RPCCW.
"A", "B", and "D" are plausible, since all of these heat exchangers are cooled by RPCCW.

Technical Reference(s): P&ID 121B (Rev. 21)
(Attach if not previously provided, P&ID 133B (Rev. 95)
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure, partial or complete, of the Service Water System, determine the effects on the system and on interrelated systems
Question Source: Bank #402440
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4, 41.5, 41.7, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 7	Tier #	1	1
K/A Statement: Anticipated Transient Without Scram:	Group #	1	1
Operational implications of EOP and AOP	K/A #	EPE 29.G2.4.20	
warnings, cautions, and notes	Importance Rating	3.8	

Proposed Question:

With the plant initially at 100% power, the following sequence of events occurs over a one-minute period:

1. The D RCP HI HI VIBRATION Annunciator is received on MB4.
2. The US directs the RO to trip the Reactor, stop the "D" RCP, and enter E-0, *Reactor Trip or Safety Injection*.
3. The RO attempts to trip the Reactor from MB4, but the Reactor does NOT trip.
4. The BOP operator attempts to trip the Reactor from MB7, but the Reactor does NOT trip.
5. The BOP operator attempts to trip the Load Center Supply breakers to Load Centers 32B and 32N, but the breakers supplying Load Center 32N do NOT open.
6. The crew enters FR-S.1, *Response to Nuclear Power Generation/ATWS*.
7. The RO begins manually inserting Control Rods.
8. The BOP operator attempts to trip the Main Turbine, but the Turbine does NOT trip.

Per FR-S.1, what is/are the next required action(s) to be taken by the RO and/or BOP operators?

a) Do NOT stop the "D" RCP. Close the MSIVs and MSIV Bypass Valves
b) Do NOT stop the "D" RCP. Runback the Turbine to close the Control Valves
c) Stop the "D" RCP, and close the MSIVs and MSIV Bypass Valves
d) Stop the "D" RCP, and runback the Turbine to close the Control Valves

Proposed Answer: B

Explanation:
Since the Reactor did not trip, and the crew was not successful at tripping the Load Center 32N supply breakers, the crew is required to enter FR-S.1, *Response to Nuclear Power Generation/ATWS*.
"C" and "D" are wrong, since per the Caution prior to step 1 of FR-S.1, the crew is required NOT to trip RCPs with power still greater than 5%. "C" and "D" are plausible, since the initial directions given by the US was to trip the Reactor, stop the "D" RCP and enter E-0.
"B" is correct, and "A" wrong, since the first action in the RNO if the turbine failed to trip is to runback the turbine to close the control valves. "A" is plausible, since if the turbine fails to runback, operators are required to close the MSIVs and Bypass Valves.

Technical Reference(s): FR-S.1 (Rev. 23), Caution prior to step 1, and Steps 1 and 2
(Attach if not previously provided, OP 3353.MB4B (Rev 11), 3-7, Step 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 and cautions of FR-S.1

Question Source: Bank #410163
Question History: Last NRC Exam Millstone 3 2017 NRC Exam
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 # 8	Tier #	1	1
K/A Statement: Steam Generator Tube Rupture:	Group #	1	1
The relationship between a SGTR and the MFW system	K/A #	EPE 38.EK2.12	
Proposed Question:	Importance Rating	3.3	

A plant startup is in progress per OP 3203, *Plant Startup*, and initial conditions are as follows:

- The plant is at 18% power.
- The crew is raising Main Turbine speed in preparation for placing the Main Generator in service.
- Main Feed is being supplied by the "A" TDMFP.

The following sequence of events occurs:

1. An undiagnosed Steam Generator Tube Rupture occurs on the "A" SG.
2. The Feed Station Operator reports difficulty controlling "A" SG level.
3. "A" SG NR level is 81% and slowly rising.
4. The BOP operator reports that the Main Turbine has tripped.
5. The RO reports RCS pressure is 2150 psia and lowering.

Which of the following Main Feedwater System responses did **NOT** automatically occur?

a) The Feed CTVs (3FWS*CTV41A-D) closed.
b) The Feed Reg Bypass Valves (3FWS*LV550, 560, 570, and 580) closed.
c) The Feed Isolation Valves (3FWS*MOV35A-D) closed.
d) The "A" TDMFP (3FWS-P2A) tripped.

Proposed Answer: C

Explanation:
The P-14 SG Level Hi-Hi setpoint of 80.5% has been exceeded. On a P-14 FWI, the Main Turbine trips, the Main Feed Pumps trip, and the Feed Reg Valves, Feed Reg Bypass Valves, and the Feed CTVs receive automatic close signals.
"A" is wrong, since the Feed CTVs receive a close signal on a P-14 FWI. "A" is plausible, since FWI signals are also generated from SIS and from Reactor Trip with low Tave, and not all FWI component actuations occur on every Feed Water Isolation (FWI) signal. Also, generally, Containment Isolation Valves close on a Containment Isolation Phase A Signal.
"B" is wrong, since the Feed Reg Bypass Valves receive a close signal on a P-14 FWI. "B" is plausible, since not all Feed Water Isolation actuations occur on every FWI signal. Also, the crew will be switching to the Main Feed Reg Valves while still at low power levels (approximately 25% power) and at that point, the Feed Reg Bypass Valves will already be closed.
"C" is correct, since the Feed Isolation Valves do not automatically close on a FWI.
"D" is wrong, since the TDMFPs trip on a P-14 FWI signal. "D" is plausible, since the TDMFPs do not automatically trip on a FWI generated from a Reactor trip with low Tave.

Technical Reference(s): Functional Sheet 13 (Rev. K)
(Attach if not previously provided, P&ID 130C (Rev. 28)
including version/revision number.) OP 3203 (Rev. 36), Step 4.3.9 Note 3, and Steps 4.3.59 and 4.3.60
Tech Spec Table 3.3-4 (Amendment 217), Functional Units 5.b and 6.c

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the main feedwater system under the following… Receipt of a P-14 signal…

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 41.8
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 9	Tier #	1	1
K/A Statement: Steam Line Rupture: Ability to use available indications to evaluate system or component status	Group #	1	1
	K/A #	APE 40.G2.1.19	
Proposed Question:	Importance Rating	3.9	

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

1. A Main Steamline faults in the Main Steam Valve Building.
2. The crew successfully completes all of the actions of E-0, *Reactor Trip or Safety injection.*
3. The crew enters E-2, *Faulted Steam Generator Isolation*.
4. While checking the faulted SG isolated in E-2, the BOP reports the indication for the “B” SG Feed Reg Valve (3FWS*FCV520) has stopped indicating on MB5.

In accordance with OP 3272, *EOP User’s Guide*, is the crew required to attempt to close the valve or verify it is closed in E-2? If not, why not?

a) The crew IS required to fail the valve closed by removing power from the valve.
b) The crew IS required to fail the valve closed by isolating instrument air to the Main Steam Valve Building.
c) The crew IS required to verify the valve is closed by checking an alternate indication.
d) The crew is NOT required to recheck the valve, since the valve was previously checked closed in E-0.

Proposed Answer: D

Explanation:
"D" is correct, and "A", "B", and "C" wrong, since, per OP 3272 EOP Users Guide, an unisolable steamline break in the MSVB will disable valve position indication for valves in the MSVB after several minutes. If valve position indication is NO longer available, previous indication of valve closure (either by direct observation or use of the PPC) is acceptable for use as there is nothing to reopen the valves once closed. Alternatively, the valves may be ensured closed by removing power since the valves are fail closed. (This is not required, since valve position was previously been checked in E-0).
“A”, “B”, and “C” are plausible, since normally a step must be completed, and these actions would verify the valve is closed or fail the valve to its closed position.

Technical Reference(s): OP 3272 (Rev. 14), Attachment 2, Section 7.0, Example 3
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: Identify and isolate a Faulted Steam Generator
Question Source: Bank #407973
Question History: Last NRC Exam Millstone 3 2009 NRC Exam
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 # 10	Tier #	1	1
K/A Statement: Loss of Main Feedwater: Operational implications and/or cause and effect relationships of: RCS parameters on a complete loss of feedwater (all SGs dried out)	Group #	1	1
	K/A #	APE 54.AK1.04	
	Importance Rating	3.9	

Proposed Question:

Initial conditions:

- The plant is at 100% power.
- AMSAC is Out of Service.

The following sequence of events occurs:

1. A loss of all Main Feedwater occurs:
2. The SGs reach the LO-LO level Reactor trip setpoint, but the Reactor fails to trip.
3. The crew enters FR-S.1, *Response To Nuclear Power Generation/ATWS.*

For this specific event, why is it imperative that the main turbine is promptly tripped?

a) To prevent the RCS from exceeding its emergency pressure limit.
b) To prevent the Fuel Clad from exceeding the 2200°F ECCS Acceptance Criterion.
c) To shutdown the Reactor by allowing the RCS to heat up.
d) To prevent an uncontrolled cooldown from adding positive reactivity to the core.

Proposed Answer: A

Explanation:
On a loss of feed ATWS, RCS pressure will rapidly rise once heat transfer to the SGs drops below heat input from the Reactor.
"A" is correct, and "B", "C", and "D" wrong, since tripping the turbine extends the time to SG tube uncovery, limiting the RCS temperature rise and resulting RCS pressure spike. IF the turbine is not tripped (by either the operator or AMSAC), RCS pressure could exceed the 3200 psia emergency pressure limit.
"B" is plausible since for other ATWS events such as loss of RCS flow, the greatest challenge to the fission product barriers will be to the Fuel Clad.
"C" is plausible, since this is the basis of step 17 in FR-S.1 that directs operators to allow the RCS to heat up to shut the Reactor down; and tripping the turbine will cause a heatup.
"D" is plausible, since this is the basis of tripping the turbine during a Subcriticality Orange Path condition, in case an uncontrolled cooldown has caused a recriticality event.

Technical Reference(s): WOG Background Document (Rev. 2) for FR-S.1, step 2
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Assuming no operator-initiated recovery technique, ANALYSE the ATWS Event leading to Core Damage.

Question Source: Bank #408296
Question History: Last NRC Exam Millstone 3 2000 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.2, 41.5, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 11	Tier #	1	1
K/A Statement: Station Blackout: Knowledge of the reasons for the length of time for which the battery capacity is designed	Group #	1	1
	K/A #	EPE 55.EK3.01	
	Importance Rating	4.1	

Proposed Question:

Complete the following statement concerning the reason the Class 1E Batteries are designed to supply all associated DC loads for a period of two hours after a loss of AC power.

The two-hour capacity is based on the assumption that after two hours, ____________.

a) the Station Blackout Diesel has been started and loaded, providing power to one train of battery chargers
b) AC power is either restored or the Emergency Diesel Generators are available to energize the battery chargers
c) the plant has been stabilized in HOT SHUTDOWN while feeding the SGs from the Turbine Driven AFW Pump
d) the plant has been cooled down to the point where Reactor Coolant Pump seal failure has become unlikely

Proposed Answer: B

Explanation:
When AC power to the battery chargers is lost, the DC loads are supplied from the batteries. The batteries are designed to supply DC loads for a period of 2 hours.
"B" is correct, and "A", "C", and "D" wrong, since analysis assumes that after two hours, either AC power is restored or the emergency generators are available to energize the battery chargers.
“A” is plausible, since the SBO diesel is a backup means of restoring AC power to one emergency bus.
“C” is plausible, since the TDAFW Pump provides heat sink without AC power.
“D” is plausible, since RCP seal failure is a major concern during a loss of all AC power, and is more likely at higher temperatures.

Technical Reference(s): FSAR 8.3.2.1.2 (Rev. 35), page 8.3-57
(Attach if not previously provided,
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 associated with the 125 VDC Distribution System, including the basis for each

Question Source: Bank #401930
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5, 41.7, 41.8 and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 12	Tier #	1	1
K/A Statement: Loss of Vital Instrument AC Bus: Knowledge of the relationship with NI	Group #	1	1
	K/A #	APE 57.AK2.08	
Proposed Question:	Importance Rating	4.1	

A plant startup is in progress per OP 3203, *Plant Startup*, and initial conditions are as follows:

- Reactor power is 14%.
- Operators are raising Reactor power by manually withdrawing control rods.

VIAC-1 deenergizes.

How does the loss of VIAC affect the NIS System?

a) The Source Range Hi Flux Trip light illuminates, resulting in an immediate automatic Reactor trip.
b) The Intermediate Range Hi Current Rod Stop (C-1) light illuminates, resulting in a block of rod withdrawal.
c) The Intermediate Range Hi Current Trip light illuminates, resulting in an immediate automatic Reactor trip.
d) The Power Range Overpower Rod Stop (C-2) light illuminates, resulting in a block of rod withdrawal.

Proposed Answer: D

Explanation:
One SR, one IR, and one PR channel are powered from VIAC-1. When VIAC-1 deenergizes, All of these NIS channels deenergize.
"A" is wrong, since the SR channel is already deenergized, and will not energize until IR power is below P-6 ($5x10^{-11}$ Amps), or power is below P-10 (11% power) and operators manually reset SRNI channels. "A" is plausible, since the plant is at reduced power, and only one channel is required to cause a Reactor trip.
"B" is wrong, since the operators blocked the IR Rod Stop when power was increased to 11% [the P-10 setpoint (11% power)]. "B" is plausible, since IR Channel N35 will de-energize, causing the IR rod stop light to illuminate, and this rod block would be received if power were below 10%.
"C" is wrong, since the operators blocked the IR Hi Current trip when power was increased to 11% [the P-10 setpoint (11% power)]. "C" is plausible, since IR Channel N35 will de-energize, causing the IR Hi Current trip light to illuminate, and this trip would occur if power were below 11%.
"D" is correct, since the overpower rod block bistable will illuminate, and this is a 1 of 4 coincidence (most trips, permissives and control signals receiving 4 inputs require 2 channels to actuate).

Technical Reference(s):	Functional Sheet 3 (Rev. G), 4 (Rev. G), and 9 (Rev. H)
(Attach if not previously provided,	EE-1BF (Rev. 35), Breakers 13 and 14
including version/revision number.)	AOP 3564 (Rev. 13), Step 1
	OP 3203 (Rev. 36), Steps 4.3.9 and 4.3.10
	Tech Spec Table 2.2-1 (Amendment 242), Functional Unit 18

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 above P-6… Intermediate Range instrument failure above P-10… Power Range instrument failure in mode 1 above P-10…

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 # 13	Tier #	1	1
K/A Statement: Loss of DC Power:	Group #	1	1
Reasons for actions contained in AOPs or EOPs	K/A #	APE 58.AK3.02	
Proposed Question:	Importance Rating	4.1	

Complete the following statement about why AOP 3563, *Loss of DC Bus Power* directs the crew to trip the Reactor if Battery 2 has been lost, but not if Battery 3 has been lost.

Battery 2 supplies the ____(1)________, while Battery 3 only supplies ____(2)________.

	(1)	(2)
a)	Turbine ETS Pressure Switches	non-vital DC loads and VIAC 3
b)	Turbine ETS Pressure Switches	VIAC 3
c)	MSIVs	non-vital DC loads and VIAC 3
d)	MSIVs	VIAC 3

Proposed Answer: D

Explanation:
"D" is correct, since Battery 2 (301B-1) supplies DC Panel 16F, which supplies the MSIVs, and if the MSIVs close, SG levels will shrink out of the narrow range, resulting in a Reactor trip. Also, the only load supplied by Battery 3 (301A-2) is VIAC 3. If bus 3 is lost, the only effect should be that the DC backup to the VIAC is lost and the VIAC remains on its normal supply. If VIAC 3 does lose power, it still should not result in a Reactor trip.
"A" and "B" are wrong, since the Turbine ETS Pressure Switches are supplied by Battery 5. "A" and "B" are plausible, since loss of the ETS Pressure Switches will result in a Reactor trip, and they are powered by a DC Bus.
"C" is wrong, since there are no DC loads supplied by Battery 3. It only supplies backup power to VIAC 3. "C" is plausible, since Battery 5 and 6 provide power to non-vital DC loads.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
- AOP 3563 (Rev. 17), Step 1
- AOP 3563 (Rev. 17), Attachment B, page 12 of 18
- AOP 3563 (Rev. 17), Attachment E, pages 9 and 11 of 11
- BKG AOP 3563 (Rev. 17), Step 1
- EE-1BB (Rev. 36)
- EE-1BC (Rev. 33), BYS*PNL-2V, Breaker 13
- EE-1BN (Rev. 22), BYS*PNL16F, Breakers 39-42

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 AOP 3563)

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.5, 41.8, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 14	Tier #	1	1
K/A Statement: Loss of Instrument Air:	Group #	1	1
Ability to operate and/or monitor the restoration of	K/A #	APE 65.AA1.03	
Systems when air pressure is regained	Importance Rating	3.1	

Proposed Question:

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

1. An air leak occurs in the Instrument Air System, and IAS pressure starts decreasing at a moderate rate.
2. The crew enters AOP 3562, *Loss of Instrument Air*.
3. The crew loops through AOP 3562, taking all required actions without tripping the Reactor.
4. A PEO locally isolates the leak by isolating a portion of piping in the Waste Disposal Building.
5. The RO and BOP report some air-operated valves have repositioned due to low air pressure.
6. Per AOP 3562, the US verifies that the crew was NOT required to trip the Reactor.
7. Instrument air pressure recovers to 110 psig.
8. The plant is still at 100% power.

Now that air pressure has been restored, what actions are the operators required to take per AOP 3562?

a) Open the RCP Thermal Barrier Cooling Isolation Valves (3CCP*AOV178A-D) and the RHR HX Temperature Control Valves 3CCP*FV66A and B).
b) Close the Main Feed Pump Recirc Valves (3FWR-FV21A and B) and place the Main Feed Regulating Valves (3FWS*FCV510-540) back to Automatic.
c) Manually restart the two previously running CDS Chillers at Main Board 1.
d) Locally place the control switches for the SWT Traveling Screens to “AUTO”.

Proposed Answer: D

Explanation:
“A” is wrong, since the RCP Thermal Barrier Cooling Isolation Valves and the RHR Heat Exchanger Temperature Control Valves have lockup solenoids that causes them to fail as is on a loss of air. “A” is plausible, since these are all air operated valves.
“B” is wrong, since the crew would have been required to trip the Reactor if feed control had been lost. "B" is plausible, since the Feed Pump Recirc Valves fail open on loss of air pressure, and AOP 3562 directs the crew to restore Main Feedwater, if required.
“C” is wrong, since if the CDS Chillers tripped during the transient, they will automatically restart. “C” is plausible, since CDS valves will reposition on a loss of air, and the Chillers may trip.
“C” is wrong, since the crew would have been required to trip the Reactor if feed control had been lost. "C" is
“D” is correct, since on a loss of instrument air, the crew was directed to place the Traveling Screens in SLOW-1, since the pneumatic instruments that control the screens in AUTO may fail; and as part of the recovery actions after air pressure is restored, the crew will be directed to place screens back in automatic.

Technical Reference(s):	AOP 3562 (Rev. 17), steps 1, 7, 10, 11, 12, 15, and 18
(Attach if not previously provided,	AOP 3562 (Rev. 17), Attachment C, page 4 of 5
including version/revision number.)	P&ID 130A (Rev. 37)
	P&ID 130C (Rev. 28)
	P&ID 130D (Rev. 29)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07008 The crew will demonstrate the ability to safely operate the plant during a loss of instrument air…

Question Source:	Bank #406972	
Question History:	Last NRC Exam	Millstone 3 2007 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 # 15	Tier #	1	1
K/A Statement: Generator Voltage and Electric Grid	Group #	1	1
Disturbance: Operational implications and/or cause/effect	K/A #	APE 77.AK1.04	
relationships with declining grid frequency or voltage	Importance Rating	3.6	

Proposed Question:

Initial conditions:

- Safety Injection has actuated.
- The crew has just verified both trains of SIS have actuated per E-0, *Reactor Trip or Safety Injection*, step 4.

The BOP Operator reports grid voltage has decreased, and the following annunciators / parameters exist at MB8:

- The BUS 34C UNDERVOLTAGE annunciator is lit.
- The BUS 34D UNDERVOLTAGE annunciator is lit.
- 345 KV switchyard voltage indicates 276 KV and stable
- Bus 34C and 34D bus voltages indicate 3700 volts and stable.

Assuming switchyard voltage conditions do not change, what will be the status of the 4160-volt buses **one minute** after the bus undervoltage annunciators came in?

a) Buses 34A and B are energized from the NSST, and the Buses 34C and D are energized from a NSST.
b) Buses 34A and B are energized from the NSST, and the Buses 34C and D are energized from the Emergency Diesels.
c) Buses 34A and B are energized from the RSST, and the Buses 34C and D are energized from a RSST.
d) Buses 34A and B are energized from the RSST, and the Buses 34C and D are energized from the Emergency Diesels.

Proposed Answer: B

Explanation:
"C" and "D" are wrong, but plausible, since the sustained UV required to lockout the NSST for the 4160V buses is 70%, and since voltages is about 90% (3744 Volts), so the NSST supply breakers will not open from this signal. "C" and "D" are plausible, since if sustained voltage was <70% (2912 Volts), the NSST would lockout.
"A" is wrong, and "B" correct, since after a brownout condition <95% (3952 Volts) exists for 7.5 seconds with an SIS signal present, the bus tie breakers will trip, and the 4160V buses will attempt to transfer to the RSSTs. But this feature is interlocked with RSST voltage, and will not occur if RSST voltage is less than 97%. This causes the bus tie breakers to open, and the emergency diesels to automatically start and energize the emergency buses. "A" is plausible, since the brownout condition <95% (3952 Volts) must exist for 4.5 minutes with no SIS signal present before the bus tie breakers trip open, and the brownout has only been in effect for one minute.

Technical Reference(s):	OP 3353.MB8A (Rev. 11), 3-12, Automatic Actions
(Attach if not previously provided,	OP 3353.MB8C (Rev. 21), 3-2, Automatic Actions
including version/revision number.)	OP 3353.MB8C (Rev. 21), 4-3 Automatic Actions
	E-0 (Rev. 36), Step 4
	LSKs 24-3A (Rev. 8), 24-3C (Rev. 8), and 24-3K (Rev. 11)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure of the 345 KV distribution system or a portion of the system, determine the effects on the system and on the interrelated systems

Question Source: Modified Bank #402128

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 41.8

Comments:
This question is considered “Modified”, since the stem has been changed to make an SIS signal present, which changes the response of the system. Also, the question asks for the status of the system after one minute, not 5 minutes. This would change the condition of the system if SIS were still not present. Also, the distractors have been changed to include which transformer is powering the normal buses, rather than state if they are energized or deenergized.

Original Bank Question #402128

With the plant operating at 100%, the following conditions exist:

- A "brown-out" condition occurs on the grid.
- 6.9KV and 4160V Bus voltages drop to 80% of normal voltage.
- This condition lasts for 5 minutes.

Assuming no operator actions have been taken, complete the following statement about the status of the 4.16KV Buses after the five minutes have passed.

a) The Normal 4160V buses are still energized at low voltage, and the Emergency 4160V buses are powered from the Emergency Diesels.
b) The Normal 4160V buses are still energized at low voltage, and the Emergency 4160V buses are powered from the RSSTs.
c) The Normal 4160V buses are de-energized, and the Emergency 4160V buses are powered from the Emergency Diesels.
d) The Normal 4160V buses are de-energized, and the Emergency 4160V buses are powered from the RSSTs.

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Question # 16	Tier #	1	1
K/A Statement: LOCA Outside Containment: Relationship with RCS leakage paths outside Containment	Group #	1	1
	K/A #	W E04.EK2.05	
Proposed Question:	Importance Rating	4.0	

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

1. Safety Injection actuates, and the crew enters E-0, *Reactor Trip or Safety Injection*.
2. During the performance of E-0, the RO reports the following:
3. RCS pressure: 1600 psia and slowly decreasing.
4. RHR flow: 200 gpm per pump.
5. SIH flow: 0 gpm per pump.
6. The crew enters ECA-1.2, *LOCA Outside Containment*.
7. The RO reports RCS pressure has slowly drifted down to 1400 psia.
8. The US directs the RO to close Cold Leg Injection Valve 3SIL*MV8809A.

Immediately after closing 3SIL*MV8809A, parameters indicate as follows:

- RCS pressure: 1400 psia.
- RHR flow: 500 gpm per pump.
- SIH flow: 50 gpm per pump.

Complete the following statement about the location and status of the leak.

The RCS leak is/was into the ___(1)___ system; and the LOCA ___(2)___ been isolated from the RCS.

	(1)	(2)
a)	SIH	HAS
b)	SIH	HAS NOT
c)	RHR	HAS
d)	RHR	HAS NOT

Proposed Answer: C

Explanation:
"A" and "B" are wrong, since the leak was into the RHR system via an unseated check valve, as indicated by RHR flow with the RCS above RHS shutoff head. "A" and "B" are plausible, since the SIH system is also susceptible to a LOCA outside Ctmt, with its piping in the ESF Bldg, and SIH flow has changed. Note that the reason SIH flow has changed is that RCS pressure has dropped below SIH Pump shutoff head. C" is correct, and "D" wrong, since RHR flow increased after the valve was closed. This indicates the leak is on the RHS side of the isolation valve, since pressure is now lower at its discharge. This indicates that the leak is now isolated from the RCS. "D" is plausible, since RHR flow is still exiting the break.

Technical Reference(s): ECA-1.2 (Rev. 10), Step 4
(Attach if not previously provided, P&ID 112A (Rev. 50)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure, partial or complete, of the Reactor Coolant System, DETERMINE the effects on the system and on interrelated systems.

Question Source: Bank #407687

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 # 17	Tier #	1	1
K/A Statement: Loss of Emergency Coolant Recirculation:	Group #	1	1
Ability to operate and/or monitor CSS	K/A #	W E11.EA1.15	
Proposed Question:	Importance Rating	3.8	

A large break LOCA has occurred, resulting in a CDA, and the following sequence of events occurs:

1. The crew aligns the RCS for cold leg recirculation per ES-1.3, *Transfer to Cold Leg Recirculation*.
2. The crew transitions back to E-1, *Loss of Reactor or Secondary Coolant*.
3. The RO reports that the "A" and "C" RSS pumps show flow and amp oscillations.
4. All other pumps' amps indicate normal.
5. The crew resets SI and CDA.

Per the E-1 foldout page, complete the following statement concerning the first two initial actions required to be taken by the crew with the RSS System.

The crew is initially required to______________________, followed by evaluating the need to stop affected ECCS pumps.

a) stop the "A" and "C" RSS Pumps while maintaining all Spray Header Isolation Valves open
b) stop the "A" and "C" RSS Pumps and close all RSS spray header isolation valves
c) stop the "C" and "D" RSS pumps while maintaining all RSS Spray Header Isolation Valves open
d) stop the "C" and "D" RSS pumps, and close all RSS Spray Header Isolation Valves

Proposed Answer: D

Explanation:
Per the E-1 Foldout Page, IF, while in CTMT Sump Recirculation, if indication of sump blockage exists as indicated by RSS or Charging or Safety Injection pumps Oscillating Amps OR Flow, the first two steps to be taken are to
1. STOP any RSS pump ONLY supplying CTMT Spray, and
2. CLOSE RSS pump Spray Header Isolation Valves (3RSS*MOV20A/B/C/D).
If sump recirc flow CANNOT be ESTABLISHED OR MAINTAINED, THEN the crew will GO TO ECA-1.1, *Loss of Emergency Coolant Recirculation*.
"A" and "B" are wrong, since the first action to be taken is to stop any RSS Pump supplying only spray flow, which are the "C" and "D" RSS Pumps. This will lower flow across the CTMT sump strainers, and hopefully stop any further cavitation. "A" and "B" are plausible, since the two pumps initially showing signs of cavitation were the "A" and "C" RSS Pumps.
"C" is wrong, and "D" correct, since after stopping the two pumps supplying only spray flow, the crew is required to close all four RSS CTMT Spray Isolation Valves. The goal is to minimize flow across the clogging CTMT sump screens while maintaining core cooling and avoid having to enter ECA-1.1 due to loss of Emergency Coolant Recirculation. "C" is plausible, since pumps have been stopped to reduce flow across the CTMT Sump strainers, and normally it is desired to maintain CTMT spray to reduce CTMT pressure.

Technical Reference(s): E-1 (Rev. 27), Foldout Page Item 7, Steps 1 and 2
(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, properly apply the notes, cautions, and foldout page items of E-1

Question Source: Bank #407662
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.6, 41.7, 41.8, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 18	Tier #	1	1
K/A Statement: Loss of Heat Sink: Knowledge of	Group #	1	1
annunciator alarms, indications, or response procedures	K/A #	W E05.G2.4.31	
Proposed Question:	Importance Rating	4.2	

An earthquake occurs, resulting in a Reactor Trip and Safety Injection, and the following initial conditions exist:

- The "B" Charging Pump failed to start, and cannot be started from MB3.
- No AFW Pumps can be started from MB5.

The crew enters FR-H.1, *Response to Loss of Secondary Heat Sink*, and the Main Board Operators make the following reports:

1. The RO reports the following:
   - The CHARG PP AUTO TRIP/OVERCURRENT annunciator has illuminated on MB3.
   - The "A" Charging pump has tripped, and cannot be started from MB3.

2. The BOP reports SG Wide Range levels indicate as follows:
   - "A" SG: 40% and slowly decreasing.
   - "B" SG: 43% and slowly decreasing.
   - "C" SG: 39% and slowly decreasing.
   - "D" SG: 39% and slowly decreasing.

Per FR-H.1, is the crew required to initiate bleed and feed cooling of the RCS? If yes, why? If not, by what method are they required to attempt to restore core cooling?

a) Yes, since no Charging Pumps are running.
b) Yes, since SG level criteria are met.
c) No. The crew will attempt to establish Main Feed flow to at least one SG.
d) No. The crew will attempt to establish Main Condensate flow to at least one SG.

Proposed Answer: A

Explanation:
Based on the initial lack of availability of the "B" Charging Pump, and the trip of the "A" Charging Pump, no Charging Pumps are running. "A" is correct, and "C" and "D" wrong, since with no charging pumps available to provide the "feed" portion of bleed and feed cooling, feed capability is significantly limited. As the RCS heats up, saturation pressure increases, meaning the PORVS may not be able to depressurize the RCS low enough for adequate SIH flow to occur, and core uncovery may result. So FR-H.1 will direct the crew to initiate bleed and feed on loss of the charging pumps. "C" and "D" are plausible, since these are both mitigation strategies of FR-H.1, and adequate SG WR levels exist. "B" is wrong, since SG WR levels are above the 29% bleed and feed criterion. "B" is plausible, since heat sink has been lost, and all four SG levels are low and decreasing.

Technical Reference(s): FR-H.1 (Rev. 28), Steps 1-3, and 12-16
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: DESCRIBE the strategy incorporated in EOP 35 FR-H.1 to safely operate the plant under conditions which require the procedure's use.

Question Source: New
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 # 19	Tier #	1	1
K/A Statement: Continuous Rod Withdrawal: Relationship with the CRDS system or components	Group #	2	2
	K/A #	APE 1.AK2.14	
Proposed Question:	Importance Rating	3.7	

Initial conditions:

- The plant at 75% power.
- Rod Control is in Automatic.

The following sequence of events occurs:

1. Control Bank D rods start stepping in automatic.
2. The crew enters AOP 3581, *Immediate Operator Actions.*
3. Per AOP 3581, the US directs the RO to place the Rod Bank Selector Switch in "Manual".
4. The RO verifies that the rods have stopped moving.
5. The crew enters AOP 3552, *Malfunction of the Rod Drive System*.

Complete the following statement about what physically occurred when the RO placed the Rod Bank Selector Switch in Manual that caused the rods to stop moving.

Selecting "Manual" removed the ___________________ the Rod Control Logic Cabinet.

a) input to the P-A Converter from
b) input to the DC Hold Bus from
c) output from the Reactor Control Unit to
d) output from the RIL Computer to

Proposed Answer: C

Explanation:
"A" is wrong, since the P-A converter output is supplied to the Bank "D" Withdrawal Limit and the RIL computer, not the Logic Cabinet. "A" is plausible, since the P-A Converter receives input from the Rod Control Logic Cabinet.
"B" is wrong, since placing Rods on the DC Hold Bus is accomplished locally via switches on the DC Hold Bus and on the individual Power Cabinets. "B" is plausible, since rods can be placed on the Hold Bus, and this would stop rod motion.
"C" is correct, since placing the Rod Bank Selector Switch in "Manual" removes the faulty automatic mod movement output signal from the Reactor Control Unit to the Logic Cabinet and replaces it with the output of the Manual Rod Control ("In-Hold-Out") Switch.
"D" is wrong, since the RIL Computer does not input to rod motion. "D" plausible, since the RIL computer does receive input from rod motion, and signals that rods are too deeply inserted in the core.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	
	AOP 3581 (Rev. 9), step 1, and Attachment A, steps A.1-A.3
	AOP 3552 (Rev. 18), Section 2.1, and Step 1
	Functional Sheet 9 (Rev. H)
	ROD014 PowerPoint Training Lesson Plan (R7C3), slides 1 and 63
	RPI014 PowerPoint Training Lesson Plan (R5C3), slide 61

Proposed references to be provided to applicants during examination:	None
Learning Objective:	Describe the operation of the following Rod Control System controls and interlocks: A. Manual Rod Control Switch B. Bank Selector Switch…
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 and 41.10
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 20	Tier #	1	1
K/A Statement: Pzr Level Control Malfunction:	Group #	2	2
Knowledge of the reasons for false indication of PZR level	K/A #	APE 28.AK3.03	
when PORV or spray valve is open and RCS is saturated	Importance Rating	3.8	

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.

Six minutes later, conditions are as follows:

- RCS pressure is 1050 psia.
- Subcooling based on core exit thermocouples is 0°F.

What does Pressurizer 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:
"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. 3), Slide 15
(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: Bank #498288

Question History: Last NRC Exam Millstone 3 2021 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.2, 41.3, 41.5, and 41.14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 21	Tier #	1	1
K/A Statement: Loss of Intermediate Range NIS:	Group #	2	2
Determine and/or interpret conditions that allow the	K/A #	APE 33.AA2.09	
bypass of an Intermediate Range level trip switch	Importance Rating	3.5	

Proposed Question:

The following sequence of events occurs:

1. NIS Intermediate Range Channel N36 fails low.
2. The crew enters AOP 3571, *Instrument Failure Response*.

For which Reactor power level band will AOP 3571 direct the crew to place the Level Trip Switch on the IR Channel N36 Drawer in BYPASS?

a) Less than 10^{-10} Amps only
b) Between 10^{-10} Amps and 5% power only
c) Between 5% and 10% power only
d) Greater than 10% power only

Proposed Answer: D

Explanation:
"A", "B", and "C" are wrong, and "D" correct, since the crew will be directed to place the Level Trip Switch in Bypass only if power is above the P-10 setpoint. If less than P-10, thew crew will be directed to restore the IR channel to OPERABLE prior to raising power above the current power band, and contact I&C to initiate repairs. "A", "B", and "C" are plausible, since each of these power bands contains separate actions in AOP 3571, Attachment E.

Technical Reference(s): AOP 3571 (Rev. 18), Attachment E, Steps E.1 through E.10
(Attach if not previously provided, OP 3360 (Rev. 8), Section 4.2
including version/revision number.) Tech Spec Table 2.2-1 (Amendment 242), Functional Unit 18

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Nuclear Instrumentation System control and interlocks… Intermediate Range Level Trip Bypass…

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 and 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 22	Tier #	1	1
K/A Statement: Fuel Handling Incident: RO duties in the control room, such as responding to alarms from the fuel handling area, communication with fuel -handling personnel, operating systems from the control room to support fueling operations, or supporting instrumentation	Group #	2	2
	K/A #	APE 36.G2.1.44	
	Importance Rating	3.9	

Proposed Question:

The plant is initially in MODE 6, and fuel is being moved from the core to the fuel pool per OP3210B, *Refueling Operations*.

A fuel bundle is dropped in the Spent Fuel Pool.

What communications link was already required to have been in place before the start of this event per the Technical Requirements Manual?

a) Health Physics personnel were required to be in communication with Containment Refueling Station personnel.
b) Health Physics personnel were required to be in communication with Fuel Building personnel.
c) Control Room personnel were required to be in communication with Containment Refueling Station personnel.
d) Control Room personnel were required to be in communication with Fuel Building personnel.

Proposed Answer: C

Explanation:
"C" is correct, and "A", "B", and "D" wrong, since per TRM 3.9.5, during core alterations, the required communications is direct communications between the Control Room and the Refueling Station.
"A", "B", and "D" are plausible, since each of these stations are manned and important during fuel movement.

Technical Reference(s): OP 3210B (Rev. 14), step 4.1.9
(Attach if not previously provided, including version/revision number.) Technical Requirements Manual 3.9.5 (LBDCR 07-MP3-018)

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 #403364
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 # 23	Tier #	1	1
K/A Statement: Loss of Condenser Vacuum: Ability to evaluate plant performance and make operational judgements based on operating characteristics, Reactor behavior, and instrument interpretation	Group #	2	2
	K/A #	APE 51.G2.1.7	
	Importance Rating	4.4	

Proposed Question:

With the plant in HOT STANDBY, the following sequence of events occurs:

1. Condenser vacuum starts slowly decreasing.
2. The crew enters AOP 3559, *Loss of Condenser Vacuum.*
3. A PEO is dispatched to investigate.

The PEO and BOP operator report the following:

- The "A", "C", and "E" Main Circulating Water Pumps are running.
- Gland sealing steam pressure is 4.5 psig.
- One Steam Jet Air Ejector is aligned for service.
- Air Ejector atmospheric discharge flow is 0 scfm.
- Air Ejector suction manifold temperature is elevated.

Based on AOP 3559, what is the likely cause of the decreasing vacuum?

a) Insufficient circ water flow to the Condenser bays exists.
b) Insufficient gland sealing steam pressure exists.
c) Air Ejector steam supply has been lost.
d) The Air Ejector is backfiring.

Proposed Answer: D

Explanation:
"A" is wrong since one Circ Pump per bay is adequate in HOT STANDBY. "A" is plausible, since Circ Water flow is lower than normal for 100% power, so this could be the problem if the plant were at 100% power.
"B" is wrong, since the normal gland sealing steam pressure band is 3 to 5 psig, and the AOP checks for between 3 and 5 psig. Current gland sealing steam pressure within this band. "B" is plausible, since if gland sealing steam pressure was low, air inleakage through the seals could occur, and the AOP checks this pressure as part of diagnosing the cause of the decreasing vacuum.
"C" is wrong, since the air ejector suction manifold temperatures are elevated. "C" is plausible, since a loss of air ejector steam would result in condenser vacuum decreasing.
"D" is correct, since indications of backfiring include elevated SJAE suction manifold temperature and/or reduced or no flow out of SJAE atmospheric discharge.

Technical Reference(s):	AOP 3559 (Rev. 14), Steps 4 through 7
(Attach if not previously provided,	OP 3329 (Rev. 15), Precaution 3.1
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 Main Turbine, Gland Seal, and Gland Exhaust Systems.
Question Source:	Modified Bank #406912 (Parent question attached)
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.5, 41.10, and 43.5

Comments:
This question is considered “Modified”, since three pertinent conditions have changed in the stem. Gland Sealing steam pressure has been raised from 0.5 to 4.5 psig, Air Ejector suction manifold temperatures are now elevated, and Air Ejector atmospheric discharge flow has decreased from 65 to 0 scfm. These changes in parameters have changed the answer to the question, making the old correct answer a new distractor.

Original Bank Question #406912

With the plant in HOT STANDBY, the following sequence of events occurs:

1. Condenser vacuum starts slowly decreasing.
2. The crew enters AOP 3559, *Loss of Condenser Vacuum.*
3. A PEO is dispatched to investigate.

The PEO and BOP operator report the following:

- The “A”, “C”, and “E” Main Circulating Water Pumps are running.
- Gland sealing steam pressure is 0.5 psig.
- Condenser Air Ejector atmospheric discharge flow is 65 scfm.
- Air Ejector suction manifold temperatures are normal.

Based on AOP 3559, what is the likely cause of the decreasing vacuum?

a) Insufficient circ water flow to the condenser bays exists.
b) Insufficient gland sealing steam pressure exists.
c) SJAE steam supply has been lost.
d) The SJAE is backfiring.

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Question # 24	Tier #	1	1
K/A Statement: Emergency Boration:	Group #	2	2
Knowledge of the relationship between Emergency Boration and the RWST	K/A #	APE 24.AK2.08	
	Importance Rating	3.8	

Proposed Question:

The plant is at 100% power.

The crew enters AOP 3566, *Immediate Boration*.

Complete the following statement.

Per AOP 3566, the RWST to Charging Pump Suction Isolation Valves (3CHS*LCV112D and 112E) are required to be opened if__(1)___, and the minimum required flow when aligned from the RWST is__(2)___.

	(1)	(2)
a)	adequate flow cannot be obtained from the BAT Tanks	33 gpm
b)	adequate flow cannot be obtained from the BAT Tanks	100 gpm
c)	a Reactor Trip is required after aligning for immediate boration	33 gpm
d)	a Reactor Trip is required after aligning for immediate boration	100 gpm

Proposed Answer: B

Explanation:
Step 1 of AOP 3566 attempts to align a boration path from the Boric Acid Tanks (BAT) through the Emergency Boration Valve. If unsuccessful, the crew will attempt to align a path from the BAT through the gravity feed boration path.
"A" is wrong, since with the RWST aligned as the boration source, a minimum flow of 100 gpm is required since RWST boron concentration is less than that of the BAT Tanks. "A" is plausible, since 33 gpm is the minimum flow required after aligning a boration path to the RCS from the BAT Tanks.
"B" is correct, since the crew will be directed to align the suctions of the Charging Pumps to the RWST if the crew is unsuccessful at aligning the gravity feed paths from the BAT Tanks, and with the RWST aligned as the boration source, a minimum flow of 100 gpm is required since RWST boron concentration is less than that of the BAT Tanks.
"C" and "D" are wrong, since the crew is allowed to have immediate boration continue from the RWST on a Reactor trip. "C" and "D" are plausible, since additional actions are required if SIS actuates while immediate boration is in progress since SIS automatically aligns the RWST to the Charging Pump suctions, and having both the RWST and BAT Tanks aligned to the suction of the Charging Pumps at the same time is an unanalyzed condition.

Technical Reference(s): AOP 3566 (Rev. 15), Note prior to Step 3
(Attach if not previously provided, AOP 3566 (Rev. 15), Step 3.d.RNO.d.1 and d.4
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 3566

Question Source: Bank # 409444

Question History: Last NRC Exam Millstone 3 2017 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 41.8 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 25	Tier #	1	1
K/A Statement: Plant Fire on Site: Ability to operate and/or monitor a Plant fire zone panel (including detector location)	Group #	2	2
	K/A #	APE 67.AA1.09	
Proposed Question:	Importance Rating	3.2	

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

A fire alarm is received in the Control Room for the Reserve Station Transformer (RSST) Area.

1. One PEO is dispatched to the RSST Area.
2. A second PEO is dispatched to Zone Panel to report the status of the applicable zone.

Which of the following capabilities are available for the PEO to take at the Zone Panel?

a) Determine which detectors are on, disabled, or in alarm.
b) Silence the local alarm horn and reset the fire alarm.
c) Check the local RSST A Water Flow alarm to verify water flow has initiated from the Fire Protection Water System.
d) Check the status of the Deluge Valve timer to verify the Deluge Valve automatically closes after the required time.

Proposed Answer: B

Explanation:
The capabilities at the Zone Panel include the following:
1. Acknowledging the alarm by depressing the "ACK" button
2. Silencing the local alarm horn by placing the silence switch in the "SIL" position
3. Testing the ability of the Zone Module to detect a "Trouble" by simulating a disconnected detector string by placing the "TEST / DISC" switch in the "DISC" position.
4. Testing the ability of the Zone Module to detect an "Alarm" by simulating a detector closed contact by placing the "TEST / DISC" switch in the "TEST" position.

"B" is correct, since these functions are available at the Zone Panel.
"A", "C", and "D" are wrong, since these functions are not available at the Zone Panel.
"A" is plausible, since this function is available at the Fire Control Station in the Control Room.
"C" and "D" are plausible, since the RSSTs are protected by a deluge spray system.

Technical Reference(s): OP 3341D (Rev. 30), Step 4.6.1.j and k
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the function of the following FPS Zone Panel Components… Zone Test Switch… Control Modules…

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, 41.7, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 26	Tier #	1	1
K/A Statement: Loss of Ctmt Integrity: Ability to determine and/or interpret facility conditions during high Ctmt pressure and selection of appropriate procedures during abnormal and emergency operations	Group #	2	2
	K/A #	W E14.EA2.01	
	Importance Rating	3.7	

Proposed Question:

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

1. The crew is performing actions in E-1, *Loss of Reactor or Secondary Coolant*.
2. Containment pressure is 65 psia and slowly increasing.
3. Containment sump level is 16 feet and slowly increasing.

Complete the following statement, assuming all other CSF Status Trees are Green or Yellow.

The color of the Containment CSF Status Tree is___(1)___, and the crew is required to transition to___(2)___.

	(1)	(2)
a)	ORANGE	FR-Z.1, *Response to High Containment Pressure*
b)	RED	FR-Z.1, *Response to High Containment Pressure*
c)	ORANGE	FR-Z.2, *Response to Containment Flooding*
d)	RED	FR-Z.2, *Response to Containment Flooding*

Proposed Answer: B

Explanation:
"B" is correct, and "A" wrong, since with CTMT pressure greater than 60 psia, the CTMT status tree is RED, requiring the crew to transition to FR-Z.1. "A" is plausible, since if CTMT pressure were between 23 psia and 60 psia with no Quench Spray pumps running, the tree would be ORANGE, and a transition to FR-Z.1 would be required.
"C" is wrong, since the CTMT status tree is RED due to high pressure, and this is a higher priority than the CTMT high sump level ORANGE path that exists with sump level above 15.75 feet. "C" is plausible, since sump level is elevated above the ORANGE path setpoint.
"D" is wrong, since with CTMT sump level above 15.75 feet, the high sump level portion of the tree is ORANGE. "D" is plausible, since CTMT Sump level is elevated, and the CTMT Status Tree does have a RED condition.

Technical Reference(s): EOP 35 F.05 (Rev. 4), CTMT Status Tree
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: IDENTIFY plant conditions which require entry into EOP 35 FR-Z.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 # 27	Tier #	2	2
K/A Statement: Reactor Coolant Pump: Electrical power to Ctmt isolation valves for RCP cooling water	Group #	1	1
	K/A #	003.K2.04	
Proposed Question:	Importance Rating	3.4	

Which 480 Volt Motor Control Center (MCC) provides power to "B" Train RPCCW Containment Supply Isolation Valve 3CCP*MOV45B?

a) MCC 32-2G
b) MCC 32-2H
c) MCC 32-3T
d) MCC 32-1W

Proposed Answer: D

Explanation:
"D" is correct, and "A", "B", and "C" wrong, since the power supply to 3CCP*MOV45B is MCC 32-1W, breaker F6F. "A", "B", and "C" are plausible, since all of these are 480-volt Motor Control Centers.

Technical Reference(s): P&ID 121B (Rev. 21)
(Attach if not previously provided, EE-1AD (Rev. 44)
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the function and location of the following major 480 volt AC System components… 480 volt Load Centers… 480 volt MCC's…
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 # 28	Tier #	2	2
K/A Statement: Reactor Coolant Pump: Operational implications or cause and effect relationships of starting one or more RCPs under various plant conditions	Group #	1	1
	K/A #	003.K5.07	
	Importance Rating	3.5	

Proposed Question:

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

- RCS Cold Leg WR Temperatures: 180°F
- Pressurizer: Solid
- RCPs: None running
- Both trains of RHR: Aligned in the "Cooldown" Mode
- RHR pumps: "B" running, "A" stopped
- COPPs: Blocked
- SG Secondary side temperatures: 240°F

What action or failure would cause an RCS overpressure transient?

a) The crew starts the "A" Residual Heat Removal pump.
b) The crew starts the "A" Reactor Coolant Pump.
c) RCS Wide Range pressure instrument 3RCS*PT405 fails high.
d) Letdown Pressure Instrument 3CHS*PT131 fails high.

Proposed Answer: B

Explanation: "B" is correct since starting an RCP with RCS Cold Leg temperatures between 150°F and 226°F with the PZR solid and SGs greater than 50°F above RCS temperature without PORVs in service will cause an RCS overpressure transient. This is due to increasing heat transfer from the SGs to the RCS. This heatup could damage the RHR suction line relief valve bellows.
"A" is wrong, since starting the 2nd RHR pump would cause the RCS pressure to decrease slightly due to increased heat removal in the RHR heat exchanger, and PCV 131 will reposition in auto to maintain RCS pressure. "A" is plausible, since starting another RHR Pump affects heat removal from the RCS.
"C" is wrong since COPPs is blocked. "C" is plausible, since RCS wide range pressure inputs to COPPS.
"D" is wrong since the letdown pressure transmitter failing high will cause the letdown pressure control valve to stroke open to attempt to lower pressure. The valve stroking open will cause RCS pressure to decrease. "D" is plausible, since the repositioning of this valve has a significant effect on RCS pressure while the Pzr is solid.

Technical Reference(s): Tech Spec Bases for LCO 3.4.1 (LBDCR No. 14-MP3-011), page B 3/4 4-1b
(Attach if not previously provided, Tech Spec LCO 3.4.1.4.1***a (Amendment 230)
including version/revision number) OP 3201 (Rev. 37), Precaution 3.3.4.c

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the procedural precautions and limitations associated with… Starting a RCP… (As available)

Question Source: Bank #406521

Question History: Millstone 3 2017 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 29	Tier #	2	2
K/A Statement: Chemical and Volume Control:	Group #	1	1
Ability to monitor automatic operation of VCT level	K/A #	004.A3.09	
Proposed Question:	Importance Rating	3.7	

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

- The Primary Makeup system is set up for AUTO operation.
- The Boration Flow Controller (FK110) pot is set at 4.5 turns.
- The Total Flow Controller (FK111) pot is set at 5.0 turns.

A VCT auto-makeup commences, and the RO is monitoring for proper system operation.

Complete the following statement.

The flowrate through Boration Flow to Blender Control Valve (3CHS*FCV110A) is_________and the flowrate through Primary Grade Water to Blender Control Valve (3CHS*FCV111A) is__________.

	3CHS*FCV110A	3CHS*FCV111A
a)	18 gpm	62 gpm
b)	36 gpm	44 gpm
c)	18 gpm	80 gpm
d)	36 gpm	80 gpm

Proposed Answer: A

Explanation:
The ratio for the boric acid pot (FK110) is 40 Gal/10 turns.
The ratio for the total flow pot (FK111) is 160 Gal/10 turns, and is normally set to 5 for auto makeup operations. When set for automatic operation, total flow controls at 80 gpm.
"A" is correct, and "B", "C", and "D" wrong, since with the boration flow pot set to 4.5, Boric Acid flow is 4.5/10 = x/40 gpm, or 18 gpm; and the total flow pot is maintaining total flow (combined PGS and boric acid flow) at 80 gpm, so PGS flow through FCV111A is 80 gpm – 18 gpm = 62 gpm.
B" is plausible, since this would be correct if the assumed ratio for Boric Acid flow is the setting for total flow of 80 gpm, and 80 gal/10 turns x 4.5/10 = 36 gpm. And 36 gpm plus 44 gpm = 80 gpm total flow.
"C" and "D" are plausible, since if the flow sensor for FK 111 was sensing PGS flow through FCV111A, rather than combined flow, its pot setting would result in 80 gpm PGS flow.

Technical Reference(s): OP3304C (Rev. 37), Section 1.3 (Discussion section)
(Attach if not previously provided, P&ID 104A (Rev. 58)
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 below listed Primary Makeup System components… Primary Makeup Flow Control Valves…

Question Source: Bank #404335

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 # 30	Tier #	2	2
K/A Statement: Chemical and Volume Control:	Group #	1	1
Ability to explain and apply system precautions,	K/A #	004.G2.1.32	
limitations, notes, or cautions	Importance Rating	3.8	

Proposed Question:

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

- The crew has placed the "B" Charging (CHS) Pump in service and stopped the "A" Charging Pump in order to perform maintenance on the "A" Pump.
- The US is performing a pre-job brief prior to dispatching a PEO to mechanically isolate the "A" CHS Pump using OP 3304A, *Charging and Letdown*.

Complete the following statement about one item that will be discussed during the brief while covering the Precautions of OP 3304A.

The discharge valve is required to be closed prior to the suction valve to avoid___________________.

a) air binding the operating pump
b) slamming the discharge check valve
c) overpressurizing the suction line
d) draining the recirculation line

Proposed Answer: C

Explanation:
"C" is correct, since the suction piping is not rated for discharge pressure, and the running pump is maintaining high pressure on the discharge header.
"A" and "D" are wrong, since the pump will be filled and vented prior to returning it to service. "A" and "D" are plausible, since potential air binding/draining lines is a concern when systems have been drained.
"B" is wrong, since check valve slamming is a concern when pumps in parallel are swapped, and this has already been completed. "B" is plausible, since check valve slamming is a concern when starting large pumps.

Technical Reference(s): OP 3304A (Rev. 47), Section 3, Precautions, especially Precaution 3.3.8
(Attach if not previously provided,
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 Chemical and Volume Control System
Question Source: Bank #402687
Question History: Last NRC Exam Millstone 3 2001 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 # 31	Tier #	2	2
K/A Statement: Residual Heat Removal: Ability to manually operate and/or monitor RHR heat exchanger temperature/bypass control valves in the control room	Group #	1	1
	K/A #	005.A4.02	
	Importance Rating	3.9	

Proposed Question:

The crew is cooling down the RCS using the "A" RHR Train per OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- RCS Hot Leg Temperature: 330°F
- RCS Pressure: 360 psia
- 3RHS*FK618 "RHR HDR FLOW" setting: 4,000 gpm
- RHR HX Outlet Valve 3RHS*FCV606 position: Fully open

The RO starts slowly adjusting the output of 3RHS*FK618 "RHR HDR FLOW" to 3,200 gpm.

How will the RO's actions affect flowrate through the RHR System?

	Flow through the RHR HX (3RHS*FCV606)	RHR HX Bypass Flow (3RHS*FCV618)
a)	Decreases	Remains the same
b)	Decreases	Decreases
c)	Increases	Remains the same
d)	Increases	Decreases

Proposed Answer: D

Explanation:
OP 3208 directs the RO to throttle down on total flow controller FK618 when an increased RCS cooldown rate is desired. Adjusting total RHR Header Flow via controller 3RHS*FK618 is accomplished by adjusting the position of the Heat Exchanger Bypass Valve.
"A" and "C" wrong, since lowering total flow is accomplished by throttling bypass valve 3RHS*FCV618 closed, decreasing HX bypass flow. "A" and "C" are plausible, since throttle valve 3RHS*FCV606 directly controls flow through the RHR Heat Exchanger, and "A" and "C" could be correct if the Total Flow Controller controlled total flow by adjusting FCV606 rather than controlling Bypass flow.
"D" is correct, and "B" wrong, since throttling closed on the bypass valve raises RHR pump discharge pressure, which increases flow through the RHR Heat Exchanger. Also, the RHR Recirc Valve will remain closed even though total flow has decreased, since total flow has remained above its auto-open setpoint of 772 gpm (and auto-close setpoint is 1541 gpm). "B" is plausible, since total flow is being reduced, so it would be expected that flow through each of the parallel paths would decrease as well. Also, FCV606 was fully open at the start of the adjustment, so flow cannot be increased through this path by opening FCV606.

Technical Reference(s): OP 3208 (Rev. 38), Section 4.3.13
(Attach if not previously provided, P&ID 112A (Rev. 50)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following residual heat removal system equipment controls and interlocks… RHR heat exchanger bypass flow control valves… RHR heat exchanger flow control valves…

Question Source: Bank #404619
Question History: Last NRC Exam Millstone 3 2007 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.7, 41.8, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 32	Tier #	2	2
K/A Statement: Emergency Core Cooling: Ability to predict the impacts of and use procedures to correct, control, or mitigate the consequences of ECCS valve failure mode on the ECCS System	Group #	1	1
	K/A #	006.A2.08	
	Importance Rating	3.2	

Proposed Question:

A large break loss of coolant accident occurs, all systems respond as designed, and initial conditions are as follows.

- The crew has completed aligning for Cold Leg Recirculation per ES-1.3, *Transfer to Cold Leg Recirculation*.
- The crew is holding their transition brief in preparation for returning to E-1, *Loss of Reactor or Secondary Coolant*.

An ESF Status Panel annunciator is received, and the RO reports the following for SI/CHG Pump Cross-connect Valves 3SIL*MV8804A and B.

- 3SIL*MV8804A has spuriously closed.
- 3SIL*MV8804B is still open.

What immediate effect, if any, does this valve failure have on the running ECCS pumps?

a) No immediate impact; suction is maintained to all running ECCS pumps.
b) Both running CHS pumps have lost suction.
c) Both running SIH pumps have lost suction.
d) The "A" Train CHS Pump and the "A" Train SIH Pump have lost suction.

Proposed Answer: A

Explanation:
During the recirculation phase of ECCS operations, the Charging and SIH pumps' suctions have been realigned to the discharge of Containment Recirculation Pumps 3RSS*P1A and B. In the cold leg recirculation lineup, the A" RSS pump is aligned to supply the suctions of the CHS pumps through 3SIL*MV8804A, while the "B" RSS Pump is aligned to supply the suctions of the SIH Pumps through 3SIL*MV8804B. Another part of aligning for cold leg recirculation includes cross-tying the CHS and SIH suction headers by opening 3SIH*MV8807A/B and 3SIH*MV8924.
"A" is correct, and "B", "C", and "D" wrong since the CHS and SIH suction headers are now cross-connected, so loss of one RSS path is lost, it will not cause a loss of suction to any ECCS Pumps.
"B" and "C" are plausible, since one RSS Pump is aligned to the CHS Pump suctions through 3SIH*MV8804A, and the other is aligned to the SIH Pump suctions through 3SIH*MV8804B, and normally, these two suction headers are not cross-connected to each other. These distractors would be correct depending on which valve failed closed if the suction headers were not cross-connected.
"D" is plausible, since 3SIH*MV8804A spuriously closed. This is an "A" Train valve. and these two pumps are "A" train pumps.

Technical Reference(s):	ES-1.3 (Rev. 20), Steps 3 and 4
(Attach if not previously provided,	P&ID 104D (Rev. 30)
including version/revision number.)	P&ID 112A (Rev. 50)
	P&ID 113A (Rev. 33)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure (partial or complete) of the Emergency Core Cooling 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. 41.5, 41.8, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 33	Tier #	2	2
K/A Statement: Emergency Core Cooling: Ability to predict and/or monitor changes in SCM associated with operation of the ECCS System	Group #	1	1
	K/A #	006.A1.06	
	Importance Rating	3.9	

Proposed Question:

The crew is performing the actions of ES-1.2, *Post-LOCA Cooldown and Depressurization*, and initial conditions are as follows:

- The crew has stopped the “B” CHS Pump.
- All other high head ECCS pumps are running.
- No RCPs are running.
- Pressurizer level is 30%.

The following sequence of events occurs:

1. The operators verify sufficient subcooling exists to stop one Safety Injection (SIH) pump.
2. The RO stops the "A" SIH pump.
3. Immediately after stopping the pump, the following:
   - RCS pressure has begun to decrease.
   - RCS subcooling has begun to decrease.

What action is the crew required to take in response to the decreasing RCS pressure and subcooling?

a) Manually reinitiate Safety Injection and transition to E-1, *Loss of Reactor or Secondary Coolant*.
b) Restart the "A" Safety Injection pump to stabilize RCS pressure and subcooling.
c) Monitor RCS pressure. Allow pressure to stabilize or increase prior to stopping the "B" SIH pump.
d) Verify subcooling is above the required value and stop the "B" SIH pump.

Proposed Answer: C

Explanation:
After stopping an SI pump, RCS pressure and subcooling will initially decrease, since break flow still exists and injection flow has been reduced. Pressure will decrease, lowering subcooling. This also decreases break flow and increases injection flow, until a new equilibrium pressure is reached where break flow equals injection flow. Initial required subcooling to stop the pump is chosen to ensure pressure and subcooling stabilize before subcooling drops below 32°F.
"A" and "B" are wrong, since SI reinitiation criteria does not apply until SI termination is completed. If subcooling is low, it will be restored as the RCS cooldown continues (break flow is decreasing, injection flow is increasing), and the second SIH pump will not be stopped until pressure stabilizes and subcooling is above the required value. “A” and “B” are plausible, since SI Reinitiation criteria exists, which would require increasing injection flow.
"C" is correct, and “D” wrong, since the crew is required to allow Pzr pressure to stabilize or increase before stopping another SI pump, since the criteria for stopping the next SI pump has been calculated assuming break flow and injection flow have equalized. “D” is plausible, since the step that stops SIH pumps requires a minimum amount of subcooling, but does not reference pressure trend or subcooling trend. This requires applying the note before the previous step.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	ES-1.2 (Rev. 22), Foldout Page Item 1 ES-1.2 (Rev. 22), steps 13 and 14, including Notes prior to Step 13 BKG EOP 35 ES-1.2 (Rev. 22), Basis for Note prior to Step 13 OP 3272 (Rev. 14), Section 3.6, top of page 17 (SI Reinitiation Criteria)

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-1.2

Question Source:	Bank #407904
Question History:	Last NRC Exam Millstone 3 2002 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	55.41.5, 41.10, and 41.11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 34	Tier #	2	2
K/A Statement: Pressurizer Relief Tank: Design features and/or interlocks that provide for draining the PRT	Group #	1	1
	K/A #	007.K4.05	
Proposed Question:	Importance Rating	2.6	

Initial conditions:

- The plant is in MODE 3 following a Reactor Trip.
- The US has directed the RO to pump down the PRT to the Gaseous Waste (DGS) System per OP 3301A, *Pressurizer Relief Tank and Reactor Vessel Flange Leakoff Operations*.

The RO depresses the "BLOCK AUTO AV8031" pushbutton for PRT Drain Outlet Valve, 3DGS-AV8031 on MB4.

Complete the following statement.

Selecting "Block" allows for 3DGS-AV8031 to ______________.

a) be manually opened when a CIA (Containment Isolation Phase A) is present
b) be manually opened when PRT level is below the high level setpoint
c) automatically close when a PRT low-pressure alarm is received
d) automatically close when a PRT low-level alarm is received

Proposed Answer: B

Explanation: PRT Drain Valve DGS-AV8031 is operated manually from MB4 by a pushbutton (OPEN/AUTO-CLOSE) with red and green position indicating lights. Normally, the valve may be opened from Main Board 4 whenever PRT level is >82%, which is the high-level alarm setpoint, and it will close when either the CLOSE push button is pressed or PRT level decreases to < 62%.
"B" is correct, and "A", "C", and "D" are wrong because the BLOCK feature blocks the PRT level interlock, allowing opening AV8031 with PRT level below the high level setpoint 82%, and blocks AV8031 from automatically closing when PRT level lowers to 62%. At 56% level, the PRT low-level annunciator will be received in the Control Room alerting the operators that PRT level is at the minimum level required to perform its function of quenching and cooling influent from the Pzr PORVs and Safety Valves.
3RCS-AV8031, PRT Drains Outlet Valve functions to prevent lowering PRT level below this minimum level. The Operator must take manual action to open the valve when level increases above 82%, and to override (block) the level interlocks if it is desired to drain the PRT below 62%.
"A" is plausible, since the PRT cannot be pumped to the Gaseous Drains System with a CIA signal present. Also, a pushbutton on Main Board 1 allows opening the Blowdown Containment Isolation Valves with a CIA signal present.
"C" is plausible, since PRT Vent Valve PCV469 has an auto-close feature based on PRT pressure.
"D" is plausible, since normally, PRT low level will automatically close the Drain Valve.

Technical Reference(s): OP 3353.MB4A (Rev. 10), 2-3
(Attach if not previously provided, OP 3353.MB4A (Rev. 10), 3-3
including version/revision number.) LSK-32-3A (Rev. 7)
OP 3301A (Rev. 13), Section 4.3, especially steps 4.3.2 and 4.3.3 including Notes
OP 3301A (Rev. 13), Precaution 3.6
P&IDs 102F (Rev. 28) and 107A (Rev. 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following Pressurizer Relief Tanks System controls and interlocks… Pressurizer Relief Tank Level Interlock…

Question Source: Bank #404300
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 10CFR55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 35	Tier #	2	2
K/A Statement: Component Cooling Water: Ability to monitor automatic actions associated with the CCWS that occur as a result of an ESFAS signal	Group #	1	1
	K/A #	008.A3.08	
	Importance Rating	4.1	

Proposed Question:

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

1. Safety Injection actuates.
2. CIA fails to actuate.

Which RPCCW valves received an automatic signal directly from the Safety Injection actuation?

a) Containment Cross-Connect Valves to Chilled Water (3CCP*MOV222 through 229)
b) Containment Header Cross-Connect Valves (3CCP*AOV179A/B and 180A/B)
c) Containment Header Isolation Valves (3CCP*MOV45A/B, 48 A/B and 49A/B)
d) Non-Safety Header Isolation Valves (3CCP*AOP197A/B, 10 A/B, 194 A/B and 19 A/B)

Proposed Answer: B

Explanation:
"A" is wrong, since the RPCCW Ctmt Cross-connect to Chilled Water Valves do not receive an automatic signal from an SIS actuation. “A” is plausible, since these valves receive an automatic open signal from an LOP or CIA.
“B” is correct, since the Containment Cross-Connect Valves receive an automatic close signal on SIS or Low RPCCW Surge Tank Level.
“C” is wrong, since the Containment Header Isolation Valves do not receive an automatic signal from an SIS actuation. “C” is plausible, since these valves do receive an automatic close signal from a CIB signal.
"D" is wrong, since the RPCCW Non-safety Header Isolation Valves do not receive an automatic signal from an SIS actuation. “D” is plausible, since these valves receive an automatic close signal from an LOP or CIA.

Technical Reference(s): P&ID 121A (Rev. 34)
(Attach if not previously provided, P&ID 121B (Rev. 21)
including version/revision number.) Functional Sheet 8 (Rev. K)
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… CCP to Chilled Water Cross-Connect Valves… Containment Header Isolation Valves… Containment Header Cross-Connect Valves… Non-Safety Header Isolation Valves
Question Source: Bank #402424
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4 and 41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 36	Tier #	2	2
K/A Statement: Pzr Pressure Control: Physical connections and/or cause and effect relationships between the PPC System and the CVCS System	Group #	1	1
	K/A #	010.K1.06	
	Importance Rating	3.7	

Proposed Question:

A plant cooldown is in progress per OP 3208, *Plant Cooldown*, and current conditions are as follows.

- The Pressurizer is solid.
- The crew is preparing to initiate Auxiliary Spray to continue cooling down the Pressurizer.

Complete the following statement.

The auxiliary spray line taps off __(1)____of the Regen Heat Exchanger; and after opening the Aux Spray Valve (3RCS*AV8145), the crew will align/maintain the Charging Header Loop Isolation Valves (3CHS*AV8146 and/or 8147) so that __(2)____during the cooldown of the Pzr.

	(1)	(2)
a)	upstream	one is open
b)	upstream	both are closed
c)	downstream	one is open
d)	downstream	both are closed

Proposed Answer: D

Explanation:
"A" and "B" are wrong, since the Aux Spray line taps off downstream of the Regenerative Heat Exchanger, in order to provide some preheating of auxiliary spray flow to minimize thermal shock of the Pzr Spray nozzle. "A" and "B" are plausible, since tapping off upstream of the Regenerative Heat Exchanger would provide a higher pressure for Auxiliary Spray flow.
"D" is correct, and "C" wrong, since the crew will be directed to close the in-service charging header loop isolation valve in order to provide increased driving head for auxiliary spray. "C" is plausible, since during normal operations and during use of the normal Pzr Spray Valves, one Charging Header Loop Isolation Valve is maintained open.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
OP 3208 (Rev. 38), Precaution 3.5.1, and Steps 4.4.14.a through e
P&ID 102C (Rev. 30)
P&ID 104A (Rev. 58)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the function and location of the following Pressurizer Pressure and Level Control System components… Pressurizer Auxiliary Spray Valve

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory of Fundamental Knowledge

10 CFR Part 55 Content: 55.41.41.4, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 37	Tier #	2	2
K/A Statement: Reactor Protection: Ability to predict and/or monitor changes in single and multiple channel trip indicators associated with operation of the RPS System	Group #	1	1
	K/A #	012.A1.04	
	Importance Rating	3.8	

Proposed Question:

The plant is stable at 7% power.

Which bistable status light combination on MB4 would directly initiate an automatic Reactor trip signal?

a) One lit "SG Lo-Lo Lvl" bistable light
b) Two lit "Pzr Hi Pres" bistable lights
c) Three lit "Pzr Level Hi" bistable lights
d) Four lit RCS "Loop Flow Low" bistable lights

Proposed Answer: B

Explanation:
"A" is wrong, since the coincidence for SG Lo-Lo Level Trip is 2/4. "A" is plausible, since this trip is active at this power level.
"B" is correct, since this trip is active at all power levels, and requires two channels to trip.
"C", and "D" are wrong, since these trips are blocked below P-7 (10% Reactor power).
"C", and "D" are plausible, since each of these signals would produce an automatic Reactor trip at higher power levels, and the correct coincidence is met.

Technical Reference(s): Functional Sheet 5 (Rev. K), 6 (Rev. J), and 7 (Rev. N)
(Attach if not previously provided, including version/revision number.) Tech Spec Table 2.2-1 (Amendment 242), Functional Unit 18

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following RPS controls and interlocks… Protective Interlocks…

Question Source: Bank #404942

Question History: Last NRC Exam Millstone 3 2009 NRC

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 # 38	Tier #	2	2
K/A Statement: Reactor Protection: Ability to prioritize and interpret the significance of each annunciator or alarm.	Group #	1	1
	K/A #	012.G2.4.45	
Proposed Question:	Importance Rating	4.1	

The plant is initially at 100% power with all selectable controllers on the Main Boards selected to Channel 1.

The following initial sequence of events occurs:

1. The RCS Loop "D" Thot instrument fails HIGH.
2. The crew enters AOP 3571, *Instrument Failure Response*.
3. The crew completes all initial actions of AOP 3571, including tripping bistables.

The PRESSURIZER HIGH PRESSURE annunciator is received on MB4 due to Pressurizer Pressure instrument 3RCS*PT457 failing HIGH.

Prior to any further operator actions or automatic control system responses, does the Reactor automatically trip due to the receipt of the Pressurizer High Pressure annunciator? Why or why not?

a) No. No coincidence is met for a Reactor trip.
b) Yes. The Reactor trips on OTΔT.
c) Yes. The Reactor trips on OPΔT.
d) Yes. The Reactor trips on high Pressurizer pressure.

Proposed Answer: A

Explanation:

The Thot failure corrective action trips numerous bistables, including the OTΔT and OPΔT bistables for loop "D". The coincidences for each of these trips is 2 of 4 channels in the tripped condition. When the pressure channel fails high, the high pressure trip Bistable for Pzr High Pressure will actuate for channel 4.
"A" is correct, and "D" wrong, since the high pressure trip requires 2 of 4 bistables, and the crew did not trip the high pressure bistable for the failed Thot instrument. "D" is plausible, since one Pzr High Pressure Bistable was received when PT457 failed high, numerous bistables had already been tripped by the crew, and the failed Pressure Instrument is on a different channel (Channel 3) than the failed Temperature channel (Channel 4).
"B" and "C" are wrong, since the failed high Pressurizer Pressure channel does not bring in a second bistable for any of the bistables that have already been tripped. "B" and "C" are plausible, since these bistables were tripped on the Thot failure, and Pzr pressure inputs to one of these trips (but increasing pressure raises the trip setpoint).

Technical Reference(s): AOP 3571 (Rev. 18), Attachment A, Step A.16, and Table A.2
(Attach if not previously provided, AOP 3571 (Rev. 18), Attachment B, Table B.1
including version/revision number.) Functional Sheet 5 (Rev. K)
Functional Sheet 6 (Rev. J)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following RPS controls and interlocks… Reactor Trip Signals…

Question Source: Bank #407219
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7, 41.10 and 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 39	Tier #	2	
K/A Statement: Engineered Safety Features Actuation: Physical connections and/or cause and effect relationships with HVAC for ESF equipment	Group #	1	1
	K/A #	013.K1.13	
	Importance Rating	3.2	

Proposed Question:

With the plant at 100% power, the crew manually starts the "B" SIH Pump for a surveillance run.

How, if at all, does the ESF Building HVAC System respond?

a) The Emergency HVAC System remains off; and the normal HVAC System remains in operation.
b) The associated emergency ACU starts and the associated emergency HVAC fans start only; and the normal HVAC remains in operation.
c) The associated emergency ACU starts and the associated emergency HVAC fans start only; and the normal HVAC dampers close and the normal HVAC fans stop.
d) All emergency ACUs start, and all emergency HVAC fans start; and the normal HVAC dampers close and the normal HVAC fans stop.

Proposed Answer: B

Explanation:
"A" and "D" are wrong, since the SIH Pump breaker closure sends a start signal to the respective emergency ACU and emergency fans, but the remainder of the emergency ventilation system remains off.
"A" is plausible, since an ESF Actuation Signal has not been received.
"D" is plausible, since an ECCS Pump has started.
"B" is correct, and "C" wrong, since the SIH Pump breaker closure sends a start signal to the respective emergency ACU and emergency fans, and the normal ventilation will remain in operation.
"C" is plausible, since an ECCS pump has started, and the normal ventilation system will shut down on an ESFAS Signal.

Technical Reference(s): P&ID 152A (Rev. 22)
(Attach if not previously provided, P&ID 152C (Rev. 18)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the HVQ System under the following Normal, Abnormal, and Emergency conditions:
A. Normal at power operation
B. Manual start of any specific ECCS pump located within the ESF Building
C. Receipt of an ESF Actuation Signal

Question Source: Bank #403769
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4 and 41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 40	Tier #	2	2
K/A Statement: Containment Cooling: COMPONENT (Breakers, Relays, and Disconnects): Interpreting a one-line diagram of control circuitry	Group #	1	1
	K/A #	(022) 191008 K1.06	
	Importance Rating	3.6	

Proposed Question:

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

1. The BOP attempts to start the “C” Containment Air Recirculation (CAR) Fan by placing the Control Switch for 3HVU-FN1C to START at VP1.
2. Immediately upon going to START, the Green, Amber, and Red lights for 3HVU-FN1C all indicate NOT LIT at VP1.
3. A PEO is dispatched to Load Center 32M to check the “C” CAR Fan breaker, and reports the following local indications for 3HVU-FN1C.

- The mechanical position indication flag is Green.
- The Red and Green position indication lights are NOT LIT.

What fault may exist in the breaker that is causing these indications? **(Reference provided)**

a) A UC fuse is blown.
b) A UT fuse is blown.
c) There is an open circuit in 3RCP-PNL contact 2-1 in the closing circuit.
d) There is an open circuit in 52L contact 7-8 in the trip circuit.

Proposed Answer: B

Explanation:
Based on the local green mechanical position indication flag showing, the breaker is open, even after going to START on VP1. This indicates that control power is not getting to the Closing Motor on the breaker. And with the red, green, and amber position indicating lights being not illuminated, control power is also not getting to the Trip Coil on the breaker.
“B” is correct, since the UT fuses supply both the closing circuit and the trip circuit for the breaker.
“A” and “C” are wrong, since if the problem was in the closing circuit, the indicating lights would still function.
“A” and “C” are plausible, since the breaker did not close when the operator selected START at VP1.
“D” is wrong, since an open in the trip circuit would not have prevented the breaker from closing. “D” is plausible, since an open in the trip circuit could cause the indicating lights to lose power.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	ESK-6BC (Rev. 10) LSK 22-27D (Rev. 7) P&ID 153A (Rev. 29)
Proposed references to be provided to applicants during examination:	**ESK-6BC**
Learning Objective:	Identify the following information on electrical control schematics (ESKs)… Circuit flowpaths… Circuit connections…
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 # 41	Tier #	2	2
K/A Statement: Containment Spray: Ability to manually operate and/or monitor CSS controls in the Control Room	Group #	1	1
	K/A #	026.A4.01	
Proposed Question:	Importance Rating	3.9	

The crew is responding to a medium sized RCS LOCA, and the following conditions exist:

- CDA actuated 14 minutes ago.
- CTMT pressure is 16.5 psia and decreasing.
- The crew is at E-1, *Loss of Reactor or Secondary Coolant*, step 7, "Check Stopping Ctmt Spray".
- The ADTS has determined that the crew will stop the "B" QSS Pump, and keep the "A" QSS pump running to reduce CTMT radiation levels.
- RWST level is 800,000 gallons.

Assuming no ESF Actuation signals have been reset, complete the following statements concerning the minimum actions physically required to meet the interlock to allow stopping the "B" QSS pump under the current conditions; AND the proper restoration position for "B" QSS Pump Discharge Spray Valve 3QSS*MOV34B after stopping the pump.

The RO will reset ___(1)___, and then take the "B" QSS Pump to STOP and back to AUTO.
The RO will then ___(2)___ the "B" QSS Pump Discharge Spray Valve.

	(1)	(2)
a)	CDA only	CLOSE
b)	CDA only	leave OPEN
c)	SIS and CDA	CLOSE
d)	SIS and CDA	leave OPEN

Proposed Answer: A

Explanation:
"C" and D" are wrong, since only the CDA signal needs to be reset in order to allow stopping of the "B" Pump ("C" and "D" wrong). "C" and "D" are plausible, since GA-8 will direct the crew to reset all ESF actuation signals.
"A" is correct, and "B" wrong, since the proper position for the discharge valve of a non-running QSS Pump is closed. "B" is plausible, since the normal standby position of ECCS injection valves is open.

Technical Reference(s): E-1 (Rev. 27), steps 7 and 24
(Attach if not previously provided, EOP 35 GA-8 (Rev. 2), steps 2 and 3
including version/revision number.) LSK 24-9.4F (Rev. 8), and 27-12F (Rev. 14)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe operation of the containment de-pressurization system in response to either a safety injection actuation signal or a containment de-pressurization accident actuation signal

Question Source: Bank #402589

Question History: Last NRC Exam Millstone 3 2004 NRC Exam

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 # 42	Tier #	2	2
K/A Statement: Main and Reheat Steam: Ability to monitor automatic operation of the Atmospheric relief valves	Group #	1	1
	K/A #	039.A3.03	
Proposed Question:	Importance Rating	3.7	

The plant is initially at 100% power and the following conditions exist:

- The Atmospheric Relief Valves (3MSS*PV20A-D) are selected to "Automatic" on MB5
- The Atmospheric Relief Valves (3MSS*PV20A-D) are set to their normal pressure setpoint.

The crew is required to trip the Reactor due to a loss of Condenser vacuum.

Complete the following statement concerning the indications available to the Main Board operators that the Atmospheric Relief Valves have throttled open to maintain RCS Tave on the trip.

The BOP Operator will observe both the Red and Green position indicating lights for 3MSS*PV20A-D __(1)__ at MB5, and the RO will observe RCS Tave stabilize at __(2)__.

	(1)	(2)
a)	ON	557°F
b)	OFF	557°F
c)	ON	561°F
d)	OFF	561°F

Proposed Answer: C

Explanation:
The position-indicating lights at MB5 for 3MSS*PV20A-D are signals developed directly from valve position. "B" and "D" are wrong, since the green light will indicate any time the valve is not fully open, and the red light will illuminate any time the valve is not fully closed. So when the valve is throttled, both the red and green lights will be lit. "B" and "D" are plausible, since numerous valves at Millstone 3 will only indicate green if the valve is fully closed, or red if the valve is fully open.
"A" is wrong, and "C" correct, since the atmospheric relief valve controller is normally set to 1125 psig (approximately 1140 psia) which is saturation pressure for 561°F. "A" is plausible, since 557°F is saturation pressure for the Condenser Steam Dump Valve setpoint of 1092 psig.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
OP 3316A (Rev. 20), step 4.2.14
P&ID 123B (Rev. 26)
ESK-7QE (Rev. 6)
Steam Tables

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

Learning Objective: Describe the operation of the following Main Steam System components, controls, and interlocks… Atmospheric Relief Valves (3MSS*PV20A-D)…

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, 41.7, 41.10, and 41.14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 43	Tier #	2	2
K/A Statement: Main Feedwater: The effect a loss or malfunction of MFW will have on CDS system or parameters	Group #	1	1
	K/A #	059.K3.01	
Proposed Question:	Importance Rating	3.2	

Initial conditions:

- The plant is operating normally at 100% power.
- The “B” and “C” Main Condensate Pumps are running.

"A" TDMFP Recirculation Valve 3FWR-FV21A fails fully OPEN.

Assuming the Reactor has NOT tripped, how does the plant initially respond to this failure?

a) Main Condensate Pump 3CNM-P1A automatically starts.
b) 4th Point Heater Drain Pump 3HDL-P1A trips.
c) The Feed Pump Low Suction Pressure Annunciator is received at MB5.
d) The Condensate Short Recycle Valve 3CNM-FV48 automatically opens.

Proposed Answer: C

Explanation:
“C” is correct, since the Feed Pump Recirc Valve failing open will divert 5,400 gpm from the feed header back to the Main Condenser, causing SGWLC to open the Feed Regulating Valves in response to the decrease in feed flow to the Steam Generators. The increased flow back to the Main Condenser, plus the Feed Reg Valves opening causes feed pump suction pressure and discharge pressure to drop. This further lowers feed flow to the SGs, causing the feed reg valves to stroke even further open. Raising feed flow to the SGs, plus the recirc flow back to the Main Condenser is beyond the capacity of the running Condensate Pumps, so feed header pressure drops. This causes an automatic start of the Motor Driven Main Feedwater Pump (auto-start on Low Feed Pump Discharge Pressure), which will lower feed pump suction pressure even more, bringing in the Low Feed Pump Suction Pressure annunciators on Main Board 5, which illuminates at 288 psig suction pressure. If suction pressure is less than 263 psig for 30 seconds, the affected Main Feed Pumps Trip.
“A” is wrong, since the standby Condensate Pump will not automatically start on increased condensate system flow. “A” is plausible, since the Main Condensate Pumps do have an Auto-Start feature (The standby pump auto-starts if a running Condensate Pump trips with its switch in the AUTO-AFTER-START position), and a secondary transient is in progress. Also, operators will be directed during this transient to manually start the standby Condensate Pump, which would clear the low suction pressure alarms and stabilize the plant.
“B” is wrong, since the 4th Point Heater Drain Pumps take a suction on the 4th Point Feedwater Heater shell side, and discharge to the Condensate System on the discharge of the 4th point feedwater heater, and the transient in progress does not significantly impact the shell side water levels of the feedwater heaters. “B” is plausible, since the 4th point Heater Drain Pumps do have automatic trips (Hi-Hi level in the associated string's 5th and/or 6th point heater(s) OR Low level in the pump's respective 4th point heater), and a secondary plant transient is in progress, but neither of these conditions are directly related to increased feedwater flow.
“D” is wrong, since 3CNM-FV48 (CND PP RECIRC MIN FLOW) is set to open as needed to maintain a minimum flow of 5,400 gpm Condensate flow, and during this transient, condensate flow has increased.
“D” is plausible, since FV48 is set to automatically open if needed, and a secondary plant transient is in progress.

Technical Reference(s):	OP 3353.MB5C, 3-4 (Rev. 16), Setpoint and automatic function
(Attach if not previously provided,	OP 3353.MB5C, 3-4 (Rev. 16), Corrective Actions 2 through 5
including version/revision number.)	3353.MB6A, 1-4 (Rev. 25), Automatic Function 3
	OP 3353.MB6A, 5-8 (Rev. 25), Automatic Functions
	OP 3321 (Rev. 35), step 4.3.38.b
	P&ID 125B (Rev. 29)
	P&ID 126A (Rev. 41)
	P&ID 126B (Rev. 49)
	P&ID 130A (Rev. 37)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Main Condensate and Makeup Control systems under the following normal, abnormal, and emergency conditions… Main condensate system startup… Long recycle…

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 # 44	Tier #	2	2
K/A Statement: Auxiliary Feedwater: Ability to predict the impacts of, and use procedures to correct, control, or mitigate the consequences of air-operated, solenoid-operated, or motor-operated valve failure on the AFW System	Group #	1	1
	K/A #	061.A2.07	
	Importance Rating	4.0	

Proposed Question:

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

1. An inadvertent SIS actuates
2. While performing the transition brief prior to exiting E-0, *Reactor Trip or Safety Injection*, the BOP operator reports the following:
   - The DWST Suction Valve to the “A” MDAFW Pump has spuriously closed.
   - SG NR Levels indicate as follows:
     - “A” SG: 20% and increasing
     - “B” SG: 23% and increasing
     - “C” SG: 21% and increasing
     - “D” SG: 20% and increasing

What are the minimum actions physically required to satisfy the interlocks to allow the operators to stop the “A” MDAFW Pump AND realign the “A” MDAFW Pump suction path to the CST from MB5?

a) Reset the Safety Injection Signal at Main Board 2.
b) Reset the A Train “Aux FW Reset for Lo-Lo SG Level” at MB5.
c) Reset both Trains of “Aux FW Reset for Lo-Lo SG Level” at MB5.
d) Reset SI at MB 2 and the A Train “Aux FW Reset for Lo-Lo SG Level” at MB 5.

Proposed Answer: A

Explanation:
Both SIS and SG Lo-Lo level signals must not be present to allow stopping a MDAFW Pump and align its suction to the CST, since the DWST is the safety grade path for the AFW Pumps.
“B” and “D” are wrong, since when SG levels are all above 18%, the Lo-Lo level signal automatically resets. “B” and “D” are plausible, since SG levels are well below their normal target of 50%, and all SG levels did shrink out of the narrow range on the trip, so a Lo-Lo condition did initially exist.
“A” is correct, and “C” wrong, since SIS needs to be reset to allow realigning the MDAFW Pp Suction Valves to the CST, since SIS aligns the valves to the DWST, and SIS has not been reset at this point in E-0. "C" is plausible, since the actions of E-0 have been completed up to the transition point, and there are places in E-0 and throughout the EOPs where operators will reset SIS. Also, the TDAFW Pump can be stopped with a SIS signal present.

Technical Reference(s) (Attach if not previously provided, including version/revision number.): OP 3322 (Rev. 37), Section 4.10, including Notes prior to step 4.10.1; LSK 6-2.1A (Rev. 14) 6-2.1B (Rev. 10) and 6-2.1C (Rev. 11); LSK 6-2.1G (Rev. 8) and 6-2.1H (Rev. 12)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure (partial or complete) of the Auxiliary Feedwater System, determine the effects on the system & on interrelated systems.

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.7, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 45	Tier #	2	2
K/A Statement: Auxiliary Feedwater System: Ability to monitor automatic start of the AFW System	Group #	1	1
	K/A #	061.A3.01	
Proposed Question:	Importance Rating	4.2	

Initial conditions:

- A plant startup is in progress per OP 3203, *Plant Startup*.
- Reactor power is 8%.

The following sequence of events occurs:

1. SIS actuates.
2. Offsite power is lost on the trip.
3. The “A” EDG fails to start.
4. One minute later, the BOP Operator is monitoring for proper AFW System response on MB5.

What will the BOP Operator observe?

a) Only the “B” MDAFW Pump is running, feeding SG’s 1 and 4.
b) Only the “B” MDAFW Pump is running, feeding SG’s 2 and 3.
c) The “B” MDAFW Pump is feeding SG’s 1 and 4, and the TDAFW Pump is feeding all four SGs.
d) The “B” MDAFW Pump is feeding SG’s 2 and 3, and the TDAFW Pump is feeding all four SGs.

Proposed Answer: B

Explanation:
The MDAFW Pumps get a start signal from 1 of 4 SGs with a Lo-Lo level condition. And the TDAFW Pump gets a start signal from 2 of 4 SGs with a Lo-Lo level condition present.
The MDAFW Pumps also get a start signal from the sequencer at 30 seconds after LOP with SIS. The TDAFWP starts on 2/4 levels < 18% in 2/4 SG.
“C” and “D” are wrong, since the TDAFW Pump does not get an automatic start signal from the sequencer, and at this low power level, SG shrink will be negligible. “C” and “D” are plausible, since at higher power levels, the TDAFW Pump would start, and the other two AFW Pumps do receive a start signal from the sequencer. Also, the TDAFW Pump would feed all four SGs if it had started.
“B” is correct, and “A” wrong, since MDAFW Pump “B” has automatically started on the SIS + LOP, and it supplies SGs 2 and 3. “A” is plausible, since the SGs fed by the Motor Driven AFW Pumps are not based on the usual Train A to 1 and 3, and Train B to 2 and 4.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
Functional Sheet 8 (Rev. K)
Functional Sheet 15 (Rev. L)
P&ID 130B (Rev. 49)
P&ID 130C (Rev. 28)
P&ID 130D (Rev. 29)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following Auxiliary Feedwater System component controls & interlocks… Motor Driven Auxiliary Feedwater Pumps… Turbine Driven Auxiliary Feedwater Pump…

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 # 46	Tier #	2	2
K/A Statement: AC Electrical Distribution: The effect of grounds on the AC Electrical Distribution System	Group #	1	1
	K/A #	062.K6.11	
Proposed Question:	Importance Rating	3.1	

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

1. A single phase "hard" ground occurs on 480 Volt Load Center 32A.
2. The LOAD CENTER GROUND DETECTION annunciator is received on MB8.

Complete the following statement about the effect of the ground, assuming no other fault exists and the plant operates as designed.

The Load Center 32A supply breaker from 4KV Bus 34A____(1)_____, and the local indication of a ground on the specific Load Center is provided by___(2)_____.

a) (1) trips open
   (2) three ground detection lights, with the grounded phase appearing dimmer than the other two
b) (1) trips open
   (2) a “volts to ground” meter, which indicates for each phase when the associated ground detection push button is depressed
c) (1) remains closed
   (2) three ground detection lights with the grounded phase appearing dimmer than the other two
d) (1) remains closed
   (2) a “volts to ground” meter, which indicates for each phase when the associated ground detection push button is depressed

Proposed Answer: C

Explanation:
“A” and “B” are wrong, since the system is ungrounded, meaning a ground on a single phase will not cause an overcurrent condition. “A” and “B” are plausible, since with hard grounds on two phases for a given bus, breaker overcurrent setpoints would be exceed causing the load center supply breaker to trip, isolating the faulty component. Also, if the system were a grounded system, which is the case for the output of the Main Generator, a single ground would cause an overcurrent condition.
“C” is correct, and “D” wrong, since ground detection is provided by specific alarms on an alarm panel on the back of MB4, as well as via 3 white lights mounted locally at the load center instrument and P.T. section. With no grounds on the load center, all 3 lights will have equal brightness. With a ground on the load center, 1 of the 3 lights will be dimmer indicating that phase is grounded. “D” is plausible, since this is how grounds are detected on the 125V DC busses.

Technical Reference(s):	OP 3353.MB8B (Rev. 15), 4-5
(Attach if not previously provided,	EE-1A (Rev. 29)
including version/revision number.)	EE-1X (Rev. 20)
	Training Lesson Plan 480026C (Rev. 8 Ch. 1) PowerPoint Slide 17

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure of the 480 VAC distribution system or a portion of the system, determine the effects on the system and on interrelated systems

Question Source:	Bank #402028
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: DC Electrical Distribution: Design features and/or interlocks that provide for Breaker interlocks, permissives, bypasses, and cross-ties	Group #	1	1
	K/A #	063.K4.02	
	Importance Rating	3.5	

Proposed Question:

What does the Kirk Key Interlock associated with Battery Charger 8 (301B-3) prevent tying together?

a) An "A" Train 480V MCC with a "B" Train MCC through the Swing Charger 8 AC input breakers
b) Two "B" Train 480V MCC's together through the Normal Charger 2 and Swing Charger 8 AC input breakers
c) An "A" Train DC Bus with a "B" Train DC Bus through the Swing Charger 8 DC output breakers
d) Two "B" Train DC Busses together through the Swing Charger 8 DC output breakers

Proposed Answer: D

Explanation:
Kirk key interlocks are provided for swing battery chargers 7, 8, and 9 to prevent cross-connecting the following 125 VDC buses:
Charger 7: Bus 1 (301A-1) and Bus 3 (301A-2)
Charger 8: Bus 2 (301B-1) and Bus 4 (301B-2)
Charger 9: Bus 5 (301C-1) and Bus 6 (301D-1)
"A" and "B" are wrong, since the Kirk key allows only one swing charger DC output breaker to be closed at a time, not the AC input breakers. "A" and "B" are plausible, since all chargers have AC input breakers.
"C" is wrong, and "D" correct, since this interlock prevents electrically cross-tying two "B" Train emergency DC Busses through the output of the swing charger. "C" is plausible, since the Kirk key interlocks prevent tying DC busses together, and Train separation is required to be maintained.

Technical Reference(s): EE-1BA (Rev. 31)
(Attach if not previously provided, OP 3345C (Rev. 17), Section 1.2
including version/revision number.) FSAR 8.3.2.1.2 (Rev. 35), page 8.3.57
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of 125 VDC distribution system controls and interlocks… Standby charger Kirk key interlocks
Question Source: Bank #401928
Question History: Last NRC Exam Millstone 3 2011 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55. 41.7 and 41.8
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 48	Tier #	2	2
K/A Statement: Emergency Diesel Generator: Electrical power supplies to the Fuel oil pumps.	Group #	1	1
	K/A #	064.K2.02	
Proposed Question:	Importance Rating	3.2	

The crew is responding to a loss of offsite power using ECA-0.0, *Loss of All AC Power*, and the following sequence of events occurs:

1. A PEO starts the "A" EDG using ECA-0.0, Attachment "E".
2. The EDG Output Breaker automatically closes and re-energizes Bus 34C.
3. A PEO reports that the "B" EDG can NOT be started.
4. After approximately 20 minutes of loaded operation the "DG A Day Tank Level Lo-Lo" annunciator comes in.
5. The PEO reports that the "A" Day Tank Level is 185 gallons and <u>neither</u> of the Fuel Oil Transfer Pumps (3EGF*P1A & 3EGF*P1C) are running.
6. Although power is available, attempts to start the Fuel Oil Transfer Pumps are unsuccessful.

Per OP 3346B, *Diesel Fuel Oil*, how will the crew fill the "A" EDG Fuel Oil Day Tank?

a) Obtain maintenance department assistance and align hoses from the “B” train Fuel Oil Transfer Pump
b) Place the 3EGS*PNLA control switch on MCC 32-1T-3H to “start” and start a “B” Train Fuel Oil Transfer Pump from 3EGS*PNLA.
c) Mechanically align the system, and use Kirk keys to electrically align and operate 3EGF*P1B, Fuel Oil Transfer Pump, from alternate power supply, MCC 32-1T.
d) Mechanically align the system, and use Kirk keys to electrically align and operate 3EGF*P1D, Fuel Oil Transfer Pump, from alternate power supply, MCC 32-1T

Proposed Answer: D

Explanation:
"A" is wrong since “B” Train power is not available for the "B" fuel oil transfer pump, and “A” Train power cannot be aligned at the switchgear to supply the “A” Pump. “A” is plausible, since non-emergency busses can be aligned to receive power from the opposite train.
"B" is wrong, since 3EGS*PNLA contains non-essential loads. “B” is plausible, since its control switch is on 32-1T, and it is operated from here when recovering in ECA-0.1.
"C" is wrong since 3EGF*P1B does not have an alternate power supply. “C” is plausible, since this does work for the “D” pump.
"D" is correct, since ARP MB8B, 5-3 directs the operators to align 3EGF*P1D to the alternate power supply and flowpath.

Technical Reference(s):	OP 3353.MB8B (Rev. 15), 5-3
(Attach if not previously provided,	OP 3346B (Rev. 11), Sections 4.6 and 4.8.5
including version/revision number.)	FSAR (Rev. 35), Figure 8.3-6
	P&ID 117A (Rev. 23)
	EE-1AK (Rev. 37)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following emergency diesel generator system components controls and interlocks… Fuel Oil Transfer Pumps…

Question Source:	Bank #403153
Question History:	Last NRC Exam Millstone 3 2002 NRC Exam
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 # 49	Tier #	2	2
K/A Statement: Emergency Diesel Generator: The effect that a loss or malfunction of the Emergency Diesel Generators will have on Systems controlled by the automatic sequencer	Group #	1	1
	K/A #	064.K3.01	
	Importance Rating	4.2	

Proposed Question:

Initial conditions:

- The plant is at 100% power.
- 120 Volt Vital Instrument Bus VIAC 2 is deenergized.

Safety Injection automatically actuates.

How will the Train "B" Diesel and the "B" SIH Pump automatically respond to the SIS?

a) The "B" EDG does NOT start, and the "B" SIH Pump does NOT start.
b) The "B" EDG does NOT start, but the "B" SIH Pump DOES start.
c) The "B" EDG STARTS, but the "B" SIH Pump does NOT start.
d) The "B" EDG STARTS, and the "B" SIH Pump DOES start.

Proposed Answer: A

Explanation:
"B" and "D" are wrong, since with the sequencer deenergized, the SIS signal will not be generated from the sequencer, so the associated "B" Train ESF loads will not automatically start. "B" and "D" are plausible, since the normally on a SIS, the "B" Train ESF loads will automatically start. Also, since there has been no LOP, power is still available from offsite power for the "B" SIH Pump.
"A" is correct, and "C" wrong, since with the sequencer de-energized, the SIS signal from the sequencer will not send the start signal to the "B" EDG. "C" is plausible, since normally on a SIS, the "B" EDG will automatically start. Also, if the initiating condition involved an LOP, the EDG would start even with the sequencer deenergized, since it would receive a start signal directly from undervoltage relays at the switchgear.

Technical Reference(s): AOP 3564 (Rev. 13), Caution prior to Step 1
(Attach if not previously provided, LSK-24-9.3A (Rev. 9)
including version/revision number.) LSK-24- 9.4A (Rev. 12)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure (partial or complete) of a emergency diesel load sequencer, determine the effects on the system and on interrelated systems.

Question Source: Bank #407017
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 # 50	Tier #	2	2
K/A Statement: Process Radiation Monitoring: The effect of a PRM component malfunction on the PRM System	Group #	1	1
	K/A #	073.K6.01	
Proposed Question:	Importance Rating	3.2	

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

1. The RMS TROUBLE annunciator is received on MB2B.
2. The RO reports the "Filter Step" alarm has been recorded for 3CMS*RE22 (CAR Fan Supply to RCP Cubicles).
3. Per US direction, the RO manually steps the 3CMS*RE22 filter per OP 3362, *Radiation Monitor Display and Control System*.
4. The RO reports the 3CMS*RE22 Filter Step alarm has cleared.

How did either the "Filter Step Alarm" or manually stepping the filter paper affect 3CMS*RE22?

a) The "Filter Step Alarm" caused 3CMS-RE22 to lose communication with the RMS Console while the filter step alarm was present.
b) The "Filter Step Alarm" caused 3CMS-RE22 to lose its Alert and Hi Alarm function while the filter step alarm was present.
c) Stepping the filter caused 3CMS-RE22 indication to drift downward due to a decreased Check Source response which can take ten to fifteen minutes to return to normal.
d) Stepping the filter caused 3CMS-RE22 indication drift upward while a new background calculation is performed which can take up to two hours to return to normal.

Proposed Answer: D

Explanation:
"A" is wrong, since the filter step alarm does not affect communication with the RMS console. "A" is plausible, since part of the RMS Trouble Alarm Response Procedure includes actions to check for an equipment failure alarm, which can cause numerous issues.
"B" is wrong, since the filter step alarm does not affect the Alert and Hi Alarm function. "B" is plausible, since part of the RMS Trouble Alarm Response Procedure includes actions to check for an equipment failure alarm, which can cause numerous issues.
"C" is wrong, since the effect on indication from stepping the filter is not caused by a change in Check Source Response. "C" is plausible, since part of the RMS Trouble Alarm Response Procedure includes actions to check the check source response if a check source failure caused the alarm, and operators are directed to observe if it is below the "expected check source" value.
"D" is correct, manually stepping the filter causes the monitor to carry out a process that includes a new calculation of background and it can take up to two hours for indications to return to normal.

Technical Reference(s): AOP 3573 (Rev. 28), Att. A, page 2 of 13, Subsequent Action 2
(Attach if not previously provided, OP 3362 (Rev. 23), Sections 4.13 and 4.16 (especially NOTE 1)
including version/revision number.) OP 3353.MB2B, 2-9 (Rev. 1), Corrective Actions 2.2 and 3

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure of the Radiation Monitoring System (partial or complete), determine the effects on the system and on inter-related 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, 41.11, and 43.2
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 51	Tier #	2	2
K/A Statement: Service Water: Ability to predict the impacts of, and use procedures to correct, control, or mitigate the consequences of Controller and positioner failures on the SWS System	Group #	1	1
	K/A #	076.A2.06	
	Importance Rating	3.3	

Proposed Question:

The plant is at 100% power, and initial conditions are as follows.

- The "A" and “B” SWP Pumps are selected to "Lead” and are running.
- The "C" and “D” SWP Pumps are selected to "Follow” and are in “Auto-After-Stop”.

Which condition will directly send an automatic start signal to the "C" Service Water Pump?

a) An automatic Safety Injection Actuation occurs.
b) A LOP signal occurs, and the "A" Service Water Pump fails to start.
c) A localized low SWP pressure condition exists at the TPCCW Heat Exchangers.
d) The "A" Emergency Diesel Generator receives an automatic start signal.

Proposed Answer: B

Explanation:
“B” is correct, since the Follow Pump receives an auto-start signal on a SIS, CDA, or LOP signal, AND the Lead SWP is not running after a 0.5 second time delay (to allow the lead pump breaker to close first). The Follow Service Water Pump also receives an auto-start signal on low SWP Pump discharge pressure for the affected train.
“A” is wrong, since the Follow Pump will not auto-start on an SIS signal unless the Lead Pump breaker fails to close. “A” is plausible, since an SIS signal starts the Lead Pump, and would start the Follow Pump if the Lead Pump failed to start.
“C” is wrong, is since the Follow Pump will not start on low pressure sensed a the inlet to the TPCCW Heat Exchangers. “C” is plausible, since a SWP Pump that has its MB1 control switch in AUTO-AFTER-STOP (normally the "follow" Pump) will automatically start with Service Water Header Pressure Lo-Lo at 26 psig on that Train's common discharge header (3SWP*PS27A/B) in the SWP pump cubicles, and there is an automatic action that will occur on low SWP pressure sensed at the inlet of the TPCCW Heat Exchangers (the SWP to TPCCW isolation MOV’s automatically close on low pressure at the common inlet to the TPCCW HX’s, but this does not input to the SW Pump start circuits).
“D” is wrong, since the EDG receives an auto start signal on a LOP, SIS, and CDA, but these signals do not auto start the Follow Pump unless the Lead Pump fails to start. “D” is plausible, since the Lead SWP receives an auto-start signal on an SIS, CDA, or LOP, and the Follow Pump would start on these signals fi the Lead Pump failed to start.

Technical Reference(s):	OP 3353.MB1C, 4-3 (Rev. 18), Automatic Functions 2 and 3
(Attach if not previously provided,	OP 3353.MB1C, 5-3 (Rev. 18), Automatic Function 1
including version/revision number.)	P&ID 133A (Rev. 48) and 133B (Rev. 95)
	LSK-24-9.4A (Rev. 12), Note 9
	LSK 24-9.4J (Rev. 8) and LSK 9-10H (Rev. 13)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure, partial or complete, of the Service Water System, determine the effects on the system and on interrelated systems.

Question Source:	Bank #405254
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.5 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 52	Tier #	2	2
K/A Statement: Instrument Air: Design features and/or interlocks that provide for Crossover to other pneumatic systems	Group #	1	1
	K/A #	078.K4.02	
Proposed Question:	Importance Rating	3.1	

With the plant at 100% power, an Instrument Air System leak occurs which is just beyond the capacity of the two IAS compressors to make up for the loss of air.

How do 3SAS-AOV33, "SERVICE AIR SUPPLY VALVE," and IAS-AOV14, "SERVICE TO INST AIR VALVE," automatically respond to IAS pressure?

a) 3SAS-AOV33 CLOSES and 3IAS-AOV14 OPENS at 90 psig decreasing. They remain aligned to supply IAS until they are manually realigned to their normal positions.
b) 3SAS-AOV33 CLOSES and 3IAS-AOV14 OPENS at 90 psig decreasing. They realign to their normal positions at 103 psig increasing.
c) 3SAS-AOV33 CLOSES and 3IAS-AOV14 OPENS at 85 psig decreasing. They remain aligned to supply IAS until they are manually realigned to their normal positions.
d) 3SAS-AOV33 CLOSES and 3IAS-AOV14 OPENS at 85 psig decreasing. They realign to their normal positions at 103 psig increasing.

Proposed Answer: D

Explanation:
The normal system line up allows automatic swapover to align Service Air to supply Instrument Air.
"A" and "B" are wrong, since as IAS receiver pressure decreases to 85 psig, IAS-AOV14 opens and SAS-AOV33 closes.
"A" and "B" are plausible, since at 90 psig decreasing, the standby Instrument Air Compressor starts.
"C" is wrong, and "D" is correct, since the valves will return to their normal positions when IAS pressure increases to 103 psig. "C" is plausible, since IAS System pressure is much more important to plant operation than Service Air pressure.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP 3353.IS (Rev. 5), 1-1, Automatic Functions
	OP 3353.IS (Rev. 5), 1-2, Automatic Functions
	OP 3332A (Rev. 29), Section 1.2
	LSK 12-1C (Rev. 6)
	LSK 12-2C (Rev. 8)
	Training Lesson Plan PAS078C (R8, Ch1), PowerPoint Slide 36

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following plant air system components controls and interlocks…… Service Air isolation valve… Service air to instrument air cross connect valve…

Question Source:	Bank #404129
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.4, 41.5, and 41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 53	Tier #	2	2
K/A Statement: Instrument Air System: Operational implications or cause and effect relationships of a loss of instrument air as it applies to the IAS System	Group #	1	1
	K/A #	078.K5.03	
	Importance Rating	3.9	

The crew has entered AOP 3562, *Loss of Instrument Air*, and the following sequence of events occurs:

1. A PEO is dispatched to perform AOP 3562, Attachment A, "Loss of Instrument Air Local Actions."
2. The PEO reports the AIR DRYER REACTIVATION BLOWER FAIL Annunciator is lit on the local Instrument Air Panel.
3. The US directs the PEO to place the Emergency Instrument Air Dryer (3IAS-DRY2) in service using OP 3332A, *Instrument Air System*.

What operational implication exists when the Emergency Air Dryer is in service?

a) Domestic Water cooling is required to be aligned to supply the "A" Instrument Air Compressor (3IAS-C1A).
b) The Emergency Air Dryer is required to be blown down, and the liquid cannot be discharged to a floor drain.
c) The backup Diesel Instrument Air Compressor (3IAS-C1C) cannot be placed in service.
d) The "B" Air Dryer Post Filter (3IAS-FLT2B) is required to be aligned for service.

Proposed Answer: B

Explanation:
"A" is wrong, since Domestic Water is not required to be aligned when placing the Emergency Air Dryer in service. "A" is plausible, since Domestic Water is the backup source of cooling water for the IAS Compressors.
"B" is correct, since the Emergency Air Dryer will accumulate the moisture that it removes from the compressed air, and it is required to be regularly blown down. Also, the liquid being blown down cannot be discharged to the floor drains. This salt slurry contains a sodium/magnesium chlorite mixture which is not permitted to be discharged though storm drains. It should be collected in a non-metallic container.
"C" is wrong, since the Backup Diesel Air Compressor can be placed in service with the emergency air dryer in service. "C" is plausible, since the Diesel Air Compressor has its own dedicated air dryer, and taps in to the Instrument Air System at a location downstream of the Emergency Air Dryer.
"D" is wrong, since the Air Dryer Post Filters are only required to be alternated if a high DP condition exists on the in-service Post Filter. "D" is plausible, since they are in the flowpath of the Emergency Air Dryer, and they will be required to be alternated per AOP 3562, Att. A if DP is high.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP 3332A (Rev. 29), Section 4.3, Caution prior to step 4.3.2.d
	OP 3332A (Rev. 29), Section 1.2, page 6 of 101
	AOP 3562 (Rev. 17), steps 2.b and c
	AOP 3562 (Rev. 17), Attachment A, steps A.2 and 3
	P&ID 138B (Rev. 37)
	P&ID 138D (Rev. 5)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major administrative or procedural precautions and limitations placed on operation of plant air systems, 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: 55.41.4, 41.5, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 54	Tier #	2	2
K/A Statement: Containment: Physical connections and/or cause and effect relationships between the CTMT System and the CTMT vacuum system	Group #	1	1
	K/A #	103.K1.07	
	Importance Rating	3.3	

Proposed Question:

Current conditions:

- A LOCA has occurred.
- The Containment Vacuum System is aligned to purge Hydrogen from Containment per GA-24, *Starting Containment Purge*.

How is the Containment Vacuum System currently aligned from and/or to Containment?

a) Both CTMT Vacuum Pumps are taking a suction on CTMT via their normal flowpaths.
b) The CTMT Vacuum Air Ejector is taking a suction on CTMT via its normal flowpath.
c) One CTMT Vacuum Pump is taking a suction on CTMT via its normal flowpath and the other Vacuum Pump is lined up to supply air to Containment from the Aux. Bldg.
d) The CTMT Vacuum Air Ejector is taking a suction on CTMT via its normal flowpath, and one Vacuum pump is lined up to supply air to CTMT from the Aux. Bldg.

Proposed Answer: C

Explanation:
"C" is correct, and "A" is wrong, since one Vacuum Pump is lined up to supply clean air to CTMT and the other is used as an exhaust pump via its normal path to the Gaseous Waste System.
"A" is plausible, since this how air is normally removed from CTMT by the CTMT Vacuum System, and a separate supply path could be aligned via the CTMT Purge System.
"B" and "D" are wrong, since the air ejector is not used to purge Hydrogen from CTMT. Its CTMT path is mechanically isolated from CTMT in MODES 1-4 by a locked-closed manual isolation valve, since its CTMT isolation valve is a manual valve that will not automatically close on a CTMT Isolation Phase A (CIA) signal.
"B" and "D" are plausible, since the Air Ejector is normally used to remove air from CTMT when initially establishing CTMT Vacuum with the plant shutdown. Also, a separate supply path could be aligned via the CTMT Purge System.

Technical Reference(s):	EOP 35 GA-24 (Rev. 2), steps 1-7
(Attach if not previously provided,	EOP 35 E-1 (Rev. 27), step 18.c.RNO
including version/revision number.)	OP 3313E (Rev. 10), Section 4.4
	P&ID 109B (Rev. 27)
	P&ID 148E (Rev. 23)
	P&ID 153A (Rev. 29)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Containment Vacuum System under the following normal, abnormal and emergency conditions: A. Initial drawdown of containment pressure. B. Maintenance of normal operating containment pressure. C. Performance of backup containment purge system operations.

Question Source:	Bank #402874
Question History:	Last NRC Exam Millstone 3 2009 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.4, 41.9, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 55	Tier #	2	2
K/A Statement: Control Rod Drive: Design features and/or interlocks that provide for Control bank sequence and overlap	Group #	2	2
	K/A #	001.K4.24	
Proposed Question:	Importance Rating	3.7	

Preparations are being made for a load increase per OP 3204, *At Power Operations*, and initial conditions are as follows:

- The plant is at 80% power.
- Bank D Control Rods are at 186 steps.

The following sequence of events occurs:

1. The US directs the RO to place rods in MANUAL for the up-power.
2. The RO inadvertently places the Bank Selector Switch in the CBD (Control Bank D) position.
3. Reactor power is then taken to 100%, all rods out (Bank D at 228 steps) condition.
4. Rod control is returned to AUTO, and the RO's error was NOT detected.

If NO further action is taken with respect to rod control, what will be the effect when the plant is subsequently shutdown?

a) Control Bank C rods will begin to insert <u>early</u>, with Bank D rods well above their proper setpoint
b) Control Bank C rods will begin to insert <u>late</u>, with Bank D rods well below their proper setpoint
c) The ROD CONTROL LIMIT LO LO annunciator will illuminate with Bank D rods well <u>above</u> the proper setpoint
d) The ROD CONTROL LIMIT LO LO annunciator will illuminate with Bank D rods well <u>below</u> the proper setpoint

Proposed Answer: A

Explanation:
The Bank Overlap Circuit determines when the individual banks start to insert in MANUAL or AUTO; and the P/A converter feeds the RIL computer and Bank "D" Full Withdrawal Limit C-11.
"C" and "D" are wrong, since in "Bank Select", the Bank Overlap Unit is frozen, but the P/A Converter still receives the outward demand signal. Also, the RIL setpoint is based on ΔT, which is also working correctly. "C" and "D" are plausible, since the RIL computer also tracks rod steps.
"A" is correct, and "B" wrong, since with the Bank Overlap Unit frozen during the rod withdrawal, it did not detect Bank D rod height increasing above 186 steps, so when rods are later inserted, it will improperly sense the point where Bank D reaches 115 steps early, and start driving in Bank C rods early. "B" is plausible, since the Bank Overlap Unit has not accurately tracked rod steps while in Bank Select.

Technical Reference(s): OP 3302A (Rev. 18), Precaution 3.6
(Attach if not previously provided, including version/revision number.) Functional Sheet 9 (Rev. H)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the function and location of the following Rod Control System components… Bank Overlap Unit… Bank Selector Switch…

Question Source: Bank #404806
Question History: Last NRC Exam Millstone 3 2019 NRC Exam
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 # 56	Tier #	2	2
K/A Statement: Pressurizer Level Control: Electrical power supplies Level channels and controllers	Group #	2	2
	K/A #	011.K2.03	
Proposed Question:	Importance Rating	3.3	

Initial conditions:

- The plant is in MODE 5.
- The crew is preparing to remove 480 Volt MCC 32-2U from service for planned electrical maintenance.

Per US direction, a PEO selects the Static Switch supplying VIAC 2 to "BYPASS SOURCE TO LOAD".

Which MCC is now supplying electrical power to Pressurizer Level Channel 2 (3RCS*LT460)?

a) MCC 32-1R
b) MCC 32-2R
c) MCC 32-1W
d) MCC 32-2W

Proposed Answer: D

Explanation:
Pressurizer Level Channel 2 receives power from 120 Volt Vital AC Bus VIAC 2, which normally receives power from MCC 32-2U, which is being removed from service.
"D" is correct, and "A", "B", and "C" wrong, since the alternate source for VIAC 2 is MCC 32-2W.
"A", "B", and "C" are plausible, since these are emergency-powered MCCs.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
EE-1BA (Rev. 31)
AOP 3571 (Rev. 18), Attachment C, page 6 of 7
OP 3345B (Rev. 13), Section 4.11
P&ID 102C (Rev. 30)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of 120 VAC Distribution System Controls and Interlocks… Static Transfer Switch Operation…

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 # 57	Tier #	2	2
K/A Statement: Nuclear Instrumentation: Operational implications or cause and effect relationships of Maximum disagreement allowed between channels as they apply to NIS System	Group #	2	2
	K/A #	015.K5.20	
	Importance Rating	3.5	

Proposed Question:

It is 11:00 AM on Tuesday with the plant at 70% power, when the following sequence of events occurs:

1. A "POWER RANGE CHANNEL DEVIATION" annunciator is received on Main Board 4.
2. Per ARP direction, the crew performs a Calorimetric per SP31002, *Plant Calorimetric*.

Complete the following statement.

The annunciator setpoint is a Power Range channel deviation of ≥ __(1)__; and per SP31002, the desired target band for all NIS Channels after adjustment is < __(2)__ deviation from the calorimetric.

	(1)	(2)
a)	4% between the highest and lowest channels	0.5%
b)	4% between the highest and lowest channels	2%
c)	2% between the highest channel and the average of the four channels	0.5%
d)	2% between the highest channel and the average of the four channels	2%

Proposed Answer: A

Explanation:
"C" and "D" are wrong, since the Power Range deviation alarm setpoint is greater than or equal to 4% difference between highest and lowest power range channels. "C" and "D" are plausible, since the upper (and lower) detector deviation alarms come in with 1 detector 2% greater than the average of the other upper (or lower) detectors. Also, the Calorimetric procedure requires the crew to calibrate NIS if there is a mismatch greater than 2% from the calorimetric.
"A" is correct, and "B" wrong, since the Calorimetric procedure directs the operators to calibrate the NIS channels to within 0.5% of the calorimetric. "B" is plausible, since the Tech Spec requirement for QPTR is for each detector to be within 2% of the average of the detectors. Also, if a coarse-adjust is required on off-hours, it is acceptable to get the NIS channel to within 2%, and I&C performance of coarse-adjust can be deferred until normal shift hours, but the question stem states this is Tuesday on day shift.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
- OP 3353.MB4C (Rev.25), 3-3, Setpoint and step 6
- SP 31002 (Rev. 26), Section 1.2
- SP 31002 (Rev. 26), steps 4.3.8, including Notes
- Tech Spec Definition 1.24 (Amendment 283)
- Tech Spec LCO 3.2.4 (Amendment 60), ACTION a

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the function and location… of the following… Flux Deviation/Miscellaneous Control and Indication Drawer… Comparator and Rate … Power Range Drawers… Power and Rate Indicators…

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, 41.7, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 58	Tier #	2	2
K/A Statement: Containment Purge: Effect of a loss or mal-function of Ctmt Purge System on CTMT system or parameters	Group #	2	2
	K/A #	029.K3.01	
Proposed Question:	Importance Rating	3.1	

The plant is in MODE 5, with initial conditions in Containment as follows:

- The CTMT Equipment Access Hatch is open.
- The CTMT Purge System is running in the Unfiltered Mode and per OP 3313F, *Containment Purge Air*.
- The CTMT Purge System is aligned per OP 3313F, Section 4.8, "Align the Containment Purge System with Containment Equipment Hatch Open".

Fuel Drop Radiation Monitor 3RMS*RE42 fails into high alarm.

What effect does this failure have on air flow through the CTMT Equipment Hatch?

a) Air flow out of CTMT through the CTMT Equipment Hatch decreases
b) Air flow out of CTMT through the CTMT Equipment Hatch increases
c) Air flow into CTMT through the CTMT Equipment Hatch decreases
d) Air flow into CTMT through the CTMT Equipment Hatch increases

Proposed Answer: C

Explanation:
The Containment Purge supply and exhaust paths are designed to supply approximately the same amount of air flow as is removed when an equal number of supply and exhaust fans are running.
"A" and "B" are wrong, since when the Equipment Hatch is open, one supply fan and two exhaust fans are maintained running to keep a constant inflow of air through the CTMT Equipment Hatch. . "A" and "B" are plausible, since the CTMT Purge system is in an unusual lineup, and depending on the lineup, air flow could be into or out through the CTMT Equipment Hatch.
"C" is correct, and "D" wrong, since when 3RMS*RE42 fails high, one CTMT Purge Supply and one CTMT Purge Exhaust damper isolates, and this stops all CTMT Purge System flow into and out of CTMT, so CTMT pressure will begin to equalize across the CTMT Equipment Hatch. "D" is plausible, since air flow was in through the CTMT Equipment Hatch, and only one train of RMS has failed, which for numerous systems, will have train-specific effects on the associated system rather than isolate both trains.

Technical Reference(s):	OP 3313F (Rev. 13), Precaution 3.3 and Note prior to step 4.1.7
(Attach if not previously provided,	OP 3313F (Rev. 13), Section 4.8, Notes 1 and 2 prior to Step 4.8.1
including version/revision number.)	OP 3313F (Rev. 13), Step 4.8.5
	AOP 3573 (Rev. 28), Attachment B, page 5 of 5
	P&IDs 148A (Rev. 42), and 153A (Rev. 29)
	LSK 22-1D (Rev. 7), and 22-27E (Rev. 6)

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:	55.41.7 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 59	Tier #	2	2
K/A Statement: Steam Dump System: Effect of RCS System or Component malfunction on SDS System	Group #	2	2
	K/A #	041.K6.10	
Proposed Question:	Importance Rating	3.5	

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

1. A load reject occurs.
2. The Condenser Steam Dump Valves ARM and OPEN.
3. The plant stabilizes at 80% power.
4. RCS Loop 4 Tave instrument fails HIGH.
5. The Reactor trips.

At what temperature will the RCS initially stabilize after the trip?

a) 567°F
b) 559°F
c) 557°F
d) 553°F

Proposed Answer: D

Explanation:
The Steam Dumps are maintained in the Tave mode at 100% power. On the initial transient, the Steam Dumps armed due to a C-7 load reject signal, and then modulated to control RCS temperature based on Program Tave (based on turbine impulse pressure) versus Auctioneered Hi Tave. When the Tave instrument failed high with the Steam Dumps already armed in the Tave Mode, Steam Dumps opened and remained open, since sensed Tave based on the failed channel will not decrease. On the plant trip, the steam dumps continue to attempt to lower Tave to program, which on the trip becomes 557°F on a plant trip.
“D” is correct, and “A”, “B”, and “C” are wrong, since with the Steam Dumps remaining open, actual Tave will drop until P-12 (553°F), which blocks the steam dumps. Then RCS Temperature will increase until P-12 automatically resets. Dumps will then reopen to maximum until temperature is less than 553°F. "A" is plausible, since 567°F is the temperature that the SG Safety Valves would maintain if the Condenser Dumps and Atmospheric Relief Valves were blocked by the failure.
"B" is plausible, since 559°F is the temperature at which the dumps would normally control temperature on a load reject, since the load reject controller has a 2°F deadband, which is designed to allow automatic rod control to drive rods in to lower Tave and close the steam dumps after a load reject event.
"C" is plausible, since 557°F is the temperature at which the steam dumps will attempt to maintain the RCS on a plant trip, since the plant trip controller does not have the 2°F deadband that the load reject controller has.

Technical Reference(s): Tech Spec Table 3.3-4 (Amendment 242, Functional Unit 9.b)
(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 tavg mode of operation) during the following normal, abnormal, and emergency conditions: a. Load reject b. Plant trip

Question Source: Bank #405072
Question History: Last NRC Exam N/A
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 # 60	Tier #	2	2
K/A Statement: Steam Generator: Ability to predict the impacts of and use procedures to correct, control, or mitigate the consequences of pressure/level transmitter failure on the SG System	Group #	2	2
	K/A #	035.A2.03	
	Importance Rating	3.8	

Proposed Question:

The plant is initially stable at 100% power.

"D" Main Steam pressure transmitter 3MSS*PT544 fails HIGH.

Complete the following statement.

Feed flow to the "D" SG starts to________________________.

a) decrease, until level error overcomes the steam flow/feed flow mismatch signal and restores SG level to 50%
b) increase, until level error overcomes the steam flow/feed flow mismatch signal and restores SG level to 50%
c) decrease, until the plant trips on Lo-Lo SG level
d) increase, until the plant trips on P-14 Hi-Hi SG level

Proposed Answer: B

Explanation:
This SGWLC System response has been tested on the simulator. "A" and "C" are wrong, since the failed high steam pressure channel inputs to steam flow for the affected SG, and this is combined with volumetric steam flow. This will indicate a higher steam density that actually exists, causing the steam flow instrument to fail high due to more indicated mass in the volume of flow being measured. With sensed steam flow increasing, the steam flow-feed flow mismatch signal will cause the Feed Reg Valve to start to open. "A" and "C" are plausible, since a density input to the SGWLC flow error circuit was affected.
"B" is correct, and "D" wrong, since the flow error signal is an anticipatory signal while level error is delayed to prevent adverse control effects from shrink and swell. SGWLC is a level dominant control system, so as actual level starts to increase due to the increased feed flow, level error will throttle the Feed Reg Valve in the closed direction. And the level error signal integrates, offsetting and then overpowering the flow error signal, eventually restoring level to 50%. The steam flow input to SGLC is clipped at (limited to) 133%, so initially, a maximum flow error of 33% exists. The thumbrule for level error needed to compensate for flow error is 3.3 to 1, so a 33% flow error will require about a 10% level error to compensate (verified on the simulator). And the SG Hi-Hi level Turbine Trip (P-14) signal does not actuate until 80%, which is well above the level error needed to stop the level increase. "D" is plausible, since flow error has driven the Feed Reg Valve open, and this would be correct if SGWLC was a flow error dominant system.

Technical Reference(s): Functional Sheet 13 (Rev. K) and Sheet 14 (Rev. K)
(Attach if not previously provided, P&IDs 123E (Rev. 25), and130D (Rev. 28)
including version/revision number.) Tech Spec Table 3.3-4 (Amend. 217), Functional Units 5.b and 6.c

Proposed references to be provided to applicants during examination: None

Learning Objective: Given the following failures (partial or complete) of the Steam Generator Water Level Control Systems, DETERMINE the effects on the system & on interrelated systems:
A. failure of the following transmitters (assume only the one specific transmitter discussed fails and all failures are associated with only 1 S/G)… Controlling Channel of narrow range S/G level… Controlling Channel of S/G Pressure

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, 41.7 and 41.14
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 61	Tier #	2	2
K/A Statement: Control Room Ventilation: Ability to manually operate and/or monitor initiate/reset of control room ventilation	Group #	2	2
	K/A #	050.A4.01	
Proposed Question:	Importance Rating	3.8	

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

1. A radiation release occurs from Millstone 2.
2. The US directs the crew to manually initiate a Control Building Isolation (CBI).

Complete the following statement.

CBI can be manually actuated from ___(1)___ in the Control Room; and when CBI actuates, one damper/valve the BOP operator will observe immediately changing position is the Control Room ___(2)___.

	(1)	(2)
a)	VP1 only	Emergency Air Recirc Damper 3HVC*AOD119A opens
b)	VP1 only	Air Inlet Valve 3HVC*AOV25 closes
c)	VP1 or MB2	Emergency Air Recirc Damper 3HVC*AOD119A opens
d)	VP1 or MB2	Air Inlet Valve 3HVC*AOV25 closes

Proposed Answer: C

Explanation:
"A" and "B" are wrong, since CBI can be manually actuated from either MB2 or Ventilation Panel 1 (VP1). "A" and "B" are plausible, since CBI equipment is monitored and operated from VP1.
"C" is correct, and "D" wrong, since a CBI causes the following:

- Kitchen Exhaust Fan Air Isolation valves AOV20 and AOV21 close
- Control Building Purge Exhaust Isolation valves AOV22 and AOV23 close
- Control Room Normal Supply Dampers AOD27A and AOD27B close
- The Train A Filter Fan starts and Emergency Air Recirc Dampers 3HVC*AOD119A/B open ("C" correct)
- The Train B Filter Fan starts only if the "A" Train Filter Fan fails to start.
- an open signal is sent to the normally open outside air inlet valves to ensure outside makeup air is brought in to pressurize the envelope via the filter ("D" wrong).
- The Technical Support Center (TSC) isolates. 30 minutes later, the TSC inlet path opens.

This alignment allows the operating ACU to maintain Control Room temperature and humidity while the Control Room is maintained at a slight positive pressure to control the rad release by minimizing ingress of contaminants into the Control Room to protect control room personnel. "D" is plausible, since the Air Inlet Valves receive a signal from a CBI, and closing these valves would prevent drawing in contamination through this path.

Technical Reference(s): OP 3314F (Rev. 41), Section 1.2, Precaution 3.8, and Section 4.13
(Attach if not previously provided, Functional Sheet 8 (Rev. K), Note 18
including version/revision number.) P&ID 151A (Rev. 33), including Note 21

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe operation of HVC system components, controls and interlocks… Control building ventilation dampers (HVC*AOD119A/B, *AOD179A/B/C, *MOD33A/B, *AOV20 through 26, *AOD27A/B, *AOD134)… Control building emergency filters (HVC*FLT1A/B)…

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, 41.10, and 41.11
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 62	Tier #	2	2
K/A Statement: Liquid Radwaste: Physical connections and/or cause and effect relationships between the Liquid Radwaste System and Sources of liquid wastes for LRS	Group #	2	2
	K/A #	068.K1.07	
	Importance Rating	2.8	

Proposed Question:

The plant is at 100% power with all systems aligned in their normal lineups.

From where are the Liquid Waste Drain Tanks (HLWDTs or LLWDTs) receiving drains?

a) The Auxiliary Building Sump
b) The Underdrain Sump outside the ESF Building
c) The Primary Drains Transfer Tank
d) The Turbine Building Floor Drains Sump

Proposed Answer: A

Explanation:
"A" is correct, since the Aux Building Sump is part of the DAS System, which drains to LWS.
"B" is wrong, since the Underdrain Sump drains to the Yard Storm Sewer System. "B" is plausible, since this sump collects drains from inside the Radiologically Controlled Area (RCA).
"C" is wrong, since the PDTT is part of the Gaseous Drains System, which drains to the Gaseous Waste System. "C" is plausible, since the PDTT receives drains that are potentially contaminated.
"D" is wrong, since the TB Floor Drains Sump normally drains to the yard storm drains system. "D" is plausible, if a high radiation alarm is received, this Sump will divert to the TPCCW Sump, which drains to the LWS System.

Technical Reference(s): P&ID 106A (Rev. 42)
(Attach if not previously provided, P&ID 106C (Rev. 50)
including version/revision number.) P&ID 107A (Rev. 28)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the function and location of the following… High Level Waste Drain Tanks… Low Level Waste Drain Tanks…
Question Source: Bank #403858
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55. 41.13
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 63	Tier #	2	2
K/A Statement: Area Radiation Monitoring: COMPONENT	Group #	2	2
(Sensors and Detectors): Theory and operation of ion chambers, Geiger-Muller tubes, and scintillation detectors	K/A #	(072) 191002K1.22	
	Importance Rating	2.8	

Proposed Question:

Complete the following statement about the Low Range Area Radiation Monitors at Millstone 3:

Low Range Area Radiation Monitors are designed to operate in the Geiger-Mueller region of the gas amplification curve since_______________in this detector operating region.

a) any radiation-induced ionization will produce a large detector output pulse
b) there is a dead time between each detected pulse due to secondary ionization
c) the gas amplification factor increases proportionally with the applied voltage
d) the size of an output pulse is dependent on the type of incident radiation

Proposed Answer: A

Explanation:
The area radiation monitors in the plant include high-range monitors inside Containment, mid-range monitors in areas of the plant where high radiation levels may occur, and low-range monitors where high radiation is improbable. Both high and mid-range monitors operate in the ion chamber region of the gas amplification curve, while the low range detectors operate in the Geiger-Mueller region.
“A” is correct, and “B”, “C”, and “D” wrong, since the benefit of using a Geiger-Mueller detector in low radiation areas is that the gas amplification factor is high in this type of detector.
“B” is plausible, since this is also a characteristic of a Geiger-Mueller detector, but it is not an advantage.
“C” is plausible, since this is a characteristic of a detector that operates in the proportional region.
“D” is plausible, since this is a characteristic of a detector operating in the ion chamber region.

Technical Reference(s): PWR GFS Components lesson plan (09 May 20) Chap. 7, Section VII.
(Attach if not previously provided, Radiation Monitoring System Lesson Plan (Rev. 6/2), page 9 of 77
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Explain the theory of operation of a Geiger-Mueller tube radiation detector

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 # 64	Tier #	3	3
K/A Statement: Conduct of Operations: Knowledge of shift or short-term relief turnover practices.	Group #	1	1
	K/A #	G2.1.3	
Proposed Question:	Importance Rating	3.7	

The plant is operating normally at 100% power.

In accordance with OP-AA-100, *Conduct of Operations*, in which of these situations is the Operator at the Controls (OATC) required to turnover OATC responsibility?

a) When acknowledging an alarm on the rear of Main Board 5 (MB5R).
b) When paging a PEO at the Communication Console.
c) When adjusting an alarm setpoint at the Radiation Monitoring Console.
d) When peer-checking the start of a fan at VP-1.

Proposed Answer: A

Explanation:
The OATC is required to remain in the At the Controls Area (ATCA) during normal operations.
"A" is correct, and "B", "C", and "D" wrong, since the OATC includes the Control Room, with the exceptions of behind the main boards, the SM office, the Work Control Areas, and the Kitchen and Bathroom. "B", "C", and "D" are plausible, since all four of these areas are inside the Control Room.

Technical Reference(s): OP-AA-100 (Rev. 46), Definitions 5.4.2 and 5.4.12
(Attach if not previously provided, OP-AA-100 (Rev. 46), Att. 2, Section 2.1, including the Figure
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the short-term relief requirements in OP-AA-100, Conduct of Operations

Question Source: Bank #455507
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 # 65	Tier #	3	3
K/A Statement: Conduct of Operations:	Group #	1	1
Knowledge of conservative decision-making practices	K/A #	G2.1.39	
Proposed Question:	Importance Rating	3.6	

Initial Conditions:

- The plant is at 40% power.
- Only the US, RO, and BOP are in the Control Room.

The following sequence of events occurs:

1. The Secondary Rounds PEO reports loud banging coming from the "B" 3rd Point Feedwater Heater.
2. The plant becomes unstable, and the crew enters AOP 3575, *Rapid Downpower*.
3. While the BOP operator is aligning the Main Turbine for the downpower per AOP 3575, the RO commences boration of the RCS per AOP 3575.
4. Power is reduced to 2% power based on Power Range NIS indications to allow repairs on the Feedwater Heater.

Which action/decision would VIOLATE conservative decision-making principles specified in OP-AA-300, *Reactivity Management*?

a) During the event, the US notifies the SM of the event as soon as practical after the downpower has been commenced, rather than before the downpower is commenced.
b) During the event, the US directs the RO to initiate a boration per AOP 3575 without a peer-checker, since the BOP is also performing actions in the AOP.
c) After stabilizing the plant at 2% power, the RO monitors diverse power indications, but monitors Intermediate Range NIS rather than ΔT power as the primary indication of the power trend.
d) After stabilizing the plant at 2% power, the SM decides to maintain power at 2% for 72 hours rather than shutting down the Reactor, to allow a quicker return to 100% power after repairs are complete.

Proposed Answer: D

Explanation:
This event is similar to the Surry event in 2005, where the decision was made to maintain power between 1 and 4% for an extended period of time, and power dropped below POAH. The RO was primarily monitoring Power Range NIS, which were not providing a reliable indication of Power, since they were at the bottom of their indicating range.
"A" is wrong, since it is acceptable for the US to NOTIFY the Shift Manager of emergent reactivity changes prior to the action or as soon as practicable following plant stabilization. "A" is plausible, since the SM would be notified of a planned downpower before commencing the downpower.
"B" is wrong, since upon entry into off-normal or emergency situations in which an AOP or EOP is entered, normal guidelines for peer checking reactivity manipulations are suspended for negative reactivity insertions. "B" is plausible, since peer checking all reactivity manipulations is a normal practice, and should be recommenced at the first available opportunity.
"C" is wrong, since at 2% power, the Power Range channels and ΔT power are at the bottom of their scale, so Intermediate Range NIS provides a better indication of Reactor Power. "C" is plausible, since at higher power levels, ΔT power and Power Range power provide a better indication of power, since IR NIS are on a logarithmic scale over multiple decades of power, while ΔT power and PR NIS are on a linear scale that is designed to monitor the power range.
"D" is correct, since the crew is required to avoid Reactor operation at low power levels for extended periods of time to the extent practicable. If Operations elects to maintain the unit in this condition due to planned maintenance activities or other pre-planned activities, then an ODM should be completed to assess the risk. Operations management shall ensure operating procedures contain adequate guidance on low power operation.

Technical Reference(s):	OP-AP-300 (Rev. 29), steps 3.2.8, 3.3.1, and 3.3.2
(Attach if not previously provided,	OP-AP-300 (Rev. 29), Attachment 1, step 15
including version/revision number.)	SP 31002 (Rev. 26), Attachment 9, items 3 and 4.

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the attributes for a conservative approach to plant operations

Question Source:	Bank #409094	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41.10, 43.5, and 43.6	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 66	Tier #	3	3
K/A Statement: Equipment Control:	Group #	2	2
Knowledge of tagging and clearance procedures	K/A #	G2.2.13	
Proposed Question:	Importance Rating	4.1	

Initial conditions:

- Near the end of an outage, a PEO is sent out to clear a large tagout in Containment
- ESOMS is down, so the "Non-Electronic Tagging Process" is being used per OP-AA-200, *Equipment Clearance*.

The following sequence of events occurs:

1. The PEO brings a copy of the tagout rather than the original, in order to prevent the tag sheet from becoming contaminated.
2. While clearing tags, PEO finds two of the tags are physically next to each other, and modifies the sequence of tag removal to remove them together due to ALARA concerns.
3. While clearing tags, the PEO discovers a small amount of boron crystals on one of the tags.
4. While exiting Containment, the PEO discards the contaminated tag in the rad waste barrel and marks on the tagout order copy that the tag was destroyed.
5. The PEO transfers the signatures to the original tagout document.

Did any of the PEO's actions directly violate the directions given in OP-AA-200 while clearing the tags? If so, which standard was violated?

a) No. The PEO dealt with the tagout copy, the ALARA issue, and the contaminated tag properly.
b) Yes. The PEO was required to use the original tagout sheet to document the clearing of tags as they were being removed.
c) Yes. The PEO was required to stop clearing tags and contact the Tagging Authority as soon as he discovered one of the tags was contaminated.
d) Yes. The PEO was required to remove the tags in the sequence specified on the tagout sheet.

Proposed Answer: D

Explanation:
"A" is wrong, and "D" correct, since OP-AA-200, step 3.2.13.q requires the tag clearer to clear the tags in the sequence specified on the tagout sheet. "A" is plausible, since there is no requirement to notify the tagging authority for the items mentioned in distractors "B" and "C", and the action taken for distractor "A" was taken due to ALARA concerns.
"B" is wrong, since there is no requirement in OP-AA-200, step 3.2.13 (Clearing a Tagout); Section 3.1 (General Practices); or Attachment 3 (Non-Electronic Tagging Process Instructions) that requires the tag clearer to use the original tagout sheet in the field. "B" is plausible, since a master copy of tagouts and procedure sections are use in the plant.
"C" is wrong, since there is no requirement in OP-AA-200, step 3.2.13 (Clearing a Tagout); Section 3.1 (General Practices); or Attachment 3 (Non-Electronic Tagging Process Instructions) that requires the tag clearer to notify the tagging authority when a contaminated tag is found. "C" is plausible, since a contaminated tag will prevent the PEO from returning the cleared tag to the tagging authority.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP-AA-200 (Rev. 35), step 3.2.13 (especially 3.2.13.q) OP-AA-200 (Rev. 35), Section 3.1 OP-AA-200 (Rev. 35), Attachment 3, Section 1.0, step 3 OP-AA-200 (Rev. 35), Attachment 3, Section 2.0, step 12

Proposed references to be provided to applicants during examination: None

Learning Objective: Outline the tag hanging requirements of the Equipment Clearance procedure

Question Source: Bank #374379

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 # 67	Tier #	3	3
K/A Statement: Equipment Control: Knowledge of the	Group #	2	2
process for controlling equipment configuration or status	K/A #	G2.2.14	
Proposed Question:	Importance Rating	3.9	

An equipment deficiency necessitates a change to the normal configuration of the Turbine Plant Component Cooling Water System.

What condition would **prohibit** the crew from tracking this alignment as an Alternate Plant Configuration per OP-AA-100, *Conduct of Operations*?

a) The alignment is included in OP 3330B, *Turbine Plant Component Cooling Water*.
b) The alignment will remain within the design of the Turbine Plant Cooling Water System.
c) The alignment will be a short-term alignment.
d) The alignment is a simple alignment.

Proposed Answer: A

Explanation:
Alternate Plant Configurations (APCs) are plant alignments which are designed into the system, but are not the normal operating configuration. They are used to document alignment of equipment and systems outside the scope of routine operations. Routine operation involves configuration changes conducted per approved procedures, written work instructions, engineering design changes, and equipment clearances.
"A" is correct, since APCs are used for alignments outside the scope of routine operations, and alignments covered by plant procedures are considered routine operations.
"B" is wrong, since APCs are plant alignments which are designed into the system. "B" is plausible, since it involves a requirement for APCs.
"C" is wrong, since specific time limits for APCs are not specified. "C" is plausible, since the length of time an APC is in place may require additional actions to be taken.
"D" is wrong, since an APC is intended to be a relatively simple configuration. "D" is plausible, since the complexity of the APC is evaluated per OP-AA-100, Attachment 9 to determine if engineering assistance will be required.

Technical Reference(s):	OP-AA-100 (Rev. 46), Definition 5.4.1
(Attach if not previously provided,	OP-AA-100 (Rev. 46), Attachment. 6, Note prior to step 15.1
including version/revision number.)	OP-AA-100 (Rev. 46), Attachment. 6, steps 15.1, 15.3, and 15.4
	OP-AA-100 (Rev. 46), Attachment 9

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the process associated with Alternate Plant Configurations (APCs).

Question Source:	Bank #370357
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 # 68	Tier #	3	3
K/A Statement: Radiation Control:	Group #	3	3
Ability to control radiation releases	K/A #	G2.3.11	
Proposed Question:	Importance Rating	3.8	

Current plant conditions:

- The plant is at 100% power.
- Radiation Monitor 3LWS-RE70 is FUNCTIONAL.

The crew is preparing to discharge the "B" Liquid Waste Test Tank (WTT) per OP 3335D, *Radioactive Liquid Waste System*.

The crew performs the following actions in preparation for discharging the "B" WTT:

1. A Valve Lineup of the "B" WTT is performed.
2. Radiation Monitor 3LWS-RE70 is purged.
3. The 3LWS-RE70 Alarm and Alert setpoints are checked.
4. A source check of 3LWS-RE70 is performed.

Per OP 3335D, which two of the above actions require an Independent Verification prior to discharging the tank?

a) 1 and 3
b) 1 and 4
c) 2 and 3
d) 2 and 4

Proposed Answer: A

Explanation:
"A" is correct, since an Independent Verification of the "B" WTT Valve Lineup and an Independent Verification of the liquid effluent monitor alarm and alert settings are required to be checked.
"B", "C", and "D" are wrong, since the purge of the radiation monitor and the source check of the radiation monitor are not required to be checked.
"B", "C", and "D" are plausible, since the purge and the source check of the Radiation Monitor are required to be performed.

Technical Reference(s): OP 3335D (Rev. 27), Steps 4.39.8 and 4.39.11.d, g, h, o, and q
(Attach if not previously provided,
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 LWS system, and 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: 55.41.10, 41.11 and 43.4
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 69	Tier #	3	3
K/A Statement: Emergency Procedures / Plan:	Group #	4	4
Knowledge of facility protection requirements, including fire brigade and portable firefighting equipment usage	K/A #	G2.4.26	
	Importance Rating	3.1	

Proposed Question:

Current conditions:

- A fire breaks out in the Unit 3 Turbine Building.
- The Site Fire Brigade is staffed with fully qualified Brigade Leads from each unit.

Complete the following statement listing the minimum composition of the Site Fire Brigade as specified in the TRM and in CM-AA-FPA-100, *Fire Protection/Appendix R (Fire Safe Shutdown) Program*.

In this event, the Site Fire Brigade requires a minimum of one MP3 Fire Brigade Lead and ________additional Brigade Members (including the MP2 Fire Brigade Lead).

a) three
b) four
c) five
d) six

Proposed Answer: B

Explanation:
"B" is correct, and "A", "C", and "D" wrong, since the site fire brigade requires a minimum of five members per the TRM, and CM-AA-FPA-100 specifies that this includes one brigade leader and four (additional) brigade members. When the brigade lead is not a fully qualified PEO on the affected unit, the affected unit is also required to supply an advisor who is at a minimum a fully qualified PEO. This is not required, since the stem states the brigade lead is a fully qualified PEO. "A", "C", and "D" are plausible, since each distractor is within ±2 members of the requirement.

Technical Reference(s): CM-AA-FPA-100 (Rev. 20) Attachment 1, step 3.7.1
(Attach if not previously provided, TRM (LBDCR 07-MP3-018) Section 6.2.2
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the composition of the Fire Brigade

Question Source: Bank #369910

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 # 70	Tier #	4	4
K/A Statement: COMMON Reactor Theory:	Group #	1	1
REACTOR KINETICS AND NEUTRON SOURCES - Explain the concept of subcritical multiplication	K/A #	192003 K1.01	
	Importance Rating	2.8	

Proposed Question:

The crew is performing a Reactor startup per OP 3202, *Reactor Startup*.

The RO withdraws control rods several steps while observing an initial increase in Source Range counts and SUR, and then stops withdrawing rods.

Assuming the Reactor is close to, but not yet critical, how will source range counts continue to respond after rod motion is stopped?

a) Counts will increase, plateau, and then decrease back to the original countrate that existed prior the rod withdrawal.
b) Counts will stop increasing as soon as the rod withdrawal stops, stabilizing at a higher level than the original countrate.
c) Counts will continue to increase for a period of time, and then stabilize at a higher level than the original countrate.
d) Counts will continue to increase at a constant startup rate until the Point of Adding Heat is reached.

Proposed Answer: C

Explanation:
Subcritical Multiplication is the process by which Source Neutrons make up for net losses in the neutron life cycle with Keff <1. This allows neutron counts to stabilize in the source range with Keff<1, since source neutrons (neutrons from sources other than the fission process due to the six-factor formula) add to the total neutron population, compensating for losses in the six factor formula. As criticality is approached and positive reactivity is added, it will take more neutron generations to stabilize out at the new equilibrium level in the subcritical Reactor.
"A" is wrong, since it doesn't consider the effects of subcritical multiplication, which won't allow counts to decrease as criticality is approached. "A" is plausible, since the Reactor is subcritical, and counts would decrease in a subcritical Reactor if subcritical multiplication was not occurring.
"B" is wrong, this ignores the fact that as criticality is approached, it takes several neutron generations to reach a new stable countrate. "B" is plausible, since this is how counts would respond if the Reactor were exactly critical and subcritical multiplication was not occurring.
"C" is correct, since countrate responds to the effects of increasing Keff and continued Subcritical Multiplication, which takes more neutron generations to reach the new higher stable countrate. This reveals itself with both a longer time for countrate to stabilize and a higher increase in countrate than previous rod withdrawals that added the same amount of positive reactivity.
"D" is wrong, since SUR will decrease to zero, and count will stabilize at higher value. "D" is plausible, since this is how neutrons respond to a rod withdrawal in a critical Reactor.

Technical Reference(s): GFS
(Attach if not previously provided, including version/revision number.) OP 3202 (Rev. 27), Note prior to step 4.27.10

Proposed references to be provided to applicants during examination: None

Learning Objective: Explain the concept of subcritical multiplication

Question Source: Bank #404050
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.1 and 41.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 71	Tier #	4	4
K/A Statement: COMMON Reactor Theory:	Group #	1	1
CONTROL RODS - Explain the shape of the curves for differential and integral rod worth versus position	K/A #	192005 K1.06	
	Importance Rating	2.9	

Proposed Question:

Complete the following statement about the Integral Rod Worth Curve in the RE Curve and Data Book.

The main reason the integral rod worth curve has a steeper slope near the center of the core than at the top and bottom of the core is due to the effects of _____________.

a) RCS Boron concentration
b) Axial flux distribution
c) Xenon concentration
d) Fuel temperature profile

Proposed Answer: B

Explanation:
Integral rod worth (IRW) is the change in reactivity (pcm) inserted by moving control rods form a reference position to any other rod height. The slope of the IRW curve at a given point is the differential rod worth (DRW) (pcm/step). And DRW is proportional to ratio of the neutron flux at the tip of the control rods versus the average neutron flux in the core.
"B" is correct, since Control Rods are a neutron poison, so they have a greater effect on reactivity based on the neutron flux in the vicinity of the tips of the control rods. So the DRW curve is similar to the axial neutron flux profile for the core, where there is higher flux in the center of the core than at the top or bottom of the core, where there is more leakage and lower neutron flux.
"A", "C", and "D" are wrong, since these factors have a much lower impact on DRW than neutron flux.
"A", "C", and "D" are plausible, since poison concentration and fuel temperature coefficient have an effect on the flux shape, but this effect is small compared to the flux profile's effect on Rod Worth

Technical Reference(s): GFS
(Attach if not previously provided, including version/revision number.) RE Curve and Data Book, Curve RE-D-02 (Cycle 21)

Proposed references to be provided to applicants during examination: None

Learning Objective: Explain the shape of the curves for differential and integral control rod worths versus rod position

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory of Fundamental Knowledge
10 CFR Part 55 Content: 55.41.1 and 41.2
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 72	Tier #	4	4
K/A Statement: COMMON Reactor Theory:	Group #	1	1
FISSION PRODUCT POISONS - Describe the effects of xenon concentration on flux shape and control rod patterns	K/A #	192006 K1.08	
	Importance Rating	3.4	

Proposed Question:

The core is at middle-of-life, and initial conditions are as follows:

- The reactor has been operating at 80 percent power for several weeks.
- Control Bank "D" Rods are at 223 steps withdrawn.
- AFD (Axial Flux Distribution) is on target at zero.

Reactor power is increased to 100 percent using boron dilution to control reactor coolant temperature while maintaining Control Bank "D" Rods at 223 steps withdrawn.

Complete the following statement about how AFD will respond to the up-power.

AFD will initially shift in the___________________over the next six hours.

a) negative direction, and then remain at that new negative value
b) positive direction, and then shift toward the negative direction
c) negative direction, and then shift toward the positive direction
d) positive direction, and then remain at that new positive value

Proposed Answer: C

Explanation:
"B" and "D" are wrong, since during an up-power (using dilution rather than withdrawing control rods), Tcold remains constant, and Thot increases. Moderator Temperature Coefficient is negative in middle of life conditions, so negative reactivity is added to the top half of the core, causing AFD to initially move in the negative direction since flux shifts to the bottom half of the core. "B" and "D" are plausible, since Rods have not moved, making it more difficult to determine which way flux will initially shift.
"A" is wrong, and "C" correct, since the higher initial flux in the bottom of the core causes more production of Iodine-135 in the bottom half of the core, which decays into Xenon over time. So as Xenon builds in in the bottom half of the core, flux will begin moving in the positive direction, and will peak in the top of the core about 6.5 hours later. "A" is plausible, since this would be true if Iodine production was not taken into account.

Technical Reference(s): GFS Lesson Plan PR06 (Rev. 4), PowerPoint Slide 40 (Fig. 6-7)
(Attach if not previously provided, Tech Spec Definition 1.4 (January 31l, 1986)
including version/revision number.) OP 3204 (Rev. 43), Section 4.4, especially step 4.4.1.e

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the effects that xenon concentration has on neutron flux shape

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.1, 41.5, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 73	Tier #	4	4
K/A Statement: COMMON Thermodynamics: THERMODYNAMIC PROCESS: Throttling and the Throttling Process - Determine the exit conditions for a throttling process based on the use of steam and/or water	Group #	2	2
	K/A #	193004 K1.15	
	Importance Rating	2.8	

Proposed Question:

The plant is in MODE 3, and current conditions are as follows:

- "D" Steam Generator pressure is 1085 psig (1100 psia).
- The "D" SG Atmospheric Relief Bypass Valve is cracked slightly open.

Assuming the "D" SG is releasing dry saturated steam, what is the approximate downstream temperature of the escaping steam when it initially reaches atmospheric pressure?

a) 212°F
b) 252°F
c) 292°F
d) 332°F

Proposed Answer: C

Explanation:
Throttling is essentially a constant enthalpy process in which no heat is being added to the fluid and no work is being done on the fluid.
"C" is correct, and "A", "B", and "D" wrong, since per the steam tables, the initial temperature of saturated steam at 1100 psia (1085 psig) is 556°F and an enthalpy of the saturated vapor of 1188.6 BTU/LBM. After finding this point on the saturated steam point on the Mollier diagram, the constant enthalpy line leads to the right until it reaches atmospheric pressure (14.7 psia). This shows superheated steam with approximately 80° of superheat. Saturation temperature at atmospheric pressure is 212° + 80°F superheat = 292° steam temperature.
"A", "B", and "D" are plausible, since these are 40° increments starting from saturation temperature at atmospheric pressure.

Technical Reference(s): Steam Tables, including the Mollier Diagram
(Attach if not previously provided, including version/revision number.) GFS

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

Learning Objective: Solve throttling process problems, applying the General Energy Equation

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 41.14
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 74	Tier #	4	4
K/A Statement: COMMON Thermodynamics:	Group #	2	2
HEAT TRANSFER: Core Thermal Power - Calculate core thermal power using a simplified heat balance	K/A #	193007 K1.08	
	Importance Rating	3.4	

Proposed Question:

An undiscovered error exists in an input to the calorimetric program, and current conditions are as follows:

- The Calorimetric calculation is selected to “Feed Flow Calc”.
- Operators have recently adjusted power range NIs to 100% based on the calorimetric.
- The crew is unaware that Power Range NIS Channels now indicate higher than actual Reactor power.

What calorimetric error could be causing this NIS inaccuracy?

a) The Feed Water temperature used in the heat balance calculation was 20°F higher than actual temperature.
b) The Reactor Coolant Pump heat input term was omitted from the heat balance calculation.
c) The Feed Water flow rate used in the heat balance calculation was 10% lower than actual flow rates.
d) The ambient heat loss term was omitted from the heat balance calculation.

Proposed Answer: B

Explanation:
Q Core = QSG + CVCS heat loss + Seal heat loss + Ambient Heat Loss - RCP Heat
"A" is wrong since QSG = mΔh, and artificially low feed temperature means the calorimetric will assume the SGs are having to heat up the feedwater a lesser amount to reach saturation temperature, resulting in a lower indicated QSG. “A” is plausible, since feedwater temperature inputs to the calorimetric.
"B" is correct, since having the RCP heat term inadvertently set to zero gives Reactor "credit" for the work added by the RCPs. This will cause Calorimetric to read artificially high.
"C" is wrong since QSG = mΔh, and artificially low feed flowrate means the calorimetric will assume the SGs are having to heat up less feedwater, resulting in a lower indicated QSG. This will cause Calorimetric to read artificially low. “C” is plausible, since feedwater flowrate inputs to the calorimetric.
"D" is wrong since having the ambient heat loss term inadvertently set to zero does not give Reactor "credit" for the work it does to replace ambient heat loss. This will cause Calorimetric to read artificially low. “A” is plausible, since ambient heat loss inputs to the calorimetric.

Technical Reference(s):	GFS
(Attach if not previously provided,	SP 31002 (Rev. 26), Attachment 4, terms A, C, I, P, R, FF, and HH
including version/revision number.)	

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

Learning Objective: Calculate core thermal power using a simplified heat balance

Question Source:	Bank #404062
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 75	Tier #	4	4
K/A Statement: COMMON Thermodynamics:	Group #	2	2
THERMAL HYDRAULICS: Natural Circulation – Describe means to determine whether natural circulation flow exists	K/A #	193008 K1.22	
	Importance Rating	4.2	

Proposed Question:

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

- The crew has entered ES-0.2, *Natural Circulation Cooldown*.
- The crew has been holding RCS cold leg wide range temperature stable for the past hour.
- Thot in all loops is 432°F.
- Tcold in all loops is 400°F.

Complete the following statement, assuming adequate natural circulation cooling is occurring.

SG pressures indicate __(1)____, and current Core Exit TC temperature trends are stable or __(2)____.

	(1)	(2)
a)	336 psig	lowering
b)	336 psig	rising
c)	232 psig	lowering
d)	232 psig	rising

Proposed Answer: C

Explanation:
The criteria used to verify adequate natural circulation is occurring in ES-0.1 includes the following:
RCS Subcooling: >32°F
SG Pressures: Stable or lowering
RCS Hot Leg WR Temperatures: Stable or lowering
Core Exit TCs: Stable or lowering
RCS Cold Leg WR Temperatures: Stable or lowering
"A" and "B" are wrong, since SG pressure will be at saturation pressure for Tcold temperature. "A" and "B" are plausible, since this is saturation pressure for Thot.
"C" is correct, since saturation pressure for 400°F is 247 psia - 15 psi = 232 psig; and Core Exit TC temperatures will be slowly decreasing due to lowering decay heat levels.
"D" is wrong, since Core Exit TC temperatures will be slowly decreasing due to lowering decay heat levels. "D" is plausible, since CETCs will be decreasing. "D" is plausible, since with lowering decay heat levels, the thermal driving head for natural circulation will be decreasing, so flowrate will be decreasing.

Technical Reference(s): ES-0.1 (Rev. 32), step 9.RNO
(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 means by which the operator can determine if natural circulation flow exists

Question Source: Modified Bank #408736 (Parent question attached below)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5 and 41.14

Comments:
This question is considered “Modified”, since the temperatures in the stem of the question have been changed, which changes the SG pressure in the correct answer. The SG pressures in all four distractors have been changed to match the new temperatures. Also, the question asks about the trend of Core Exit Thermocouples, rather than RCS Hot Leg temperature trend.

Original Bank Question #408376

A hurricane warning is in effect for southeastern Connecticut, and current conditions are as follows:

- A Reactor trip has occurred due to a loss of offsite power.
- The crew has cooled down the plant 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 are reading 425°F.
- Thot in all loops is 424°F.
- Tcold in all loops is 392°F.

If adequate natural circulation is occurring, what should be the approximate SG pressures, and what is the current Thot trend?

a) SG pressures reading 210 psig, with Thot slowly increasing due to decreasing natural circulation flow rates.
b) SG pressures reading 210 psig, with Thot slowly decreasing due to lowering decay heat levels.
c) SG pressures reading 307 psig, with Thot slowly increasing due to decreasing natural circulation flow rates.
d) SG pressures reading 307 psig, with Thot slowly decreasing due to lowering decay heat levels.

B

Examination Outline Cross-reference:	Level	RO	SRO
Question # 76	Tier #		1
K/A Statement: Pzr Vapor Space Accident: Ability to determine and/or interpret the effect of an open PORV or code safety based on observation of plant parameters	Group #		1
	K/A #	APE 8.AA2.20	
	Importance Rating		4.0

Proposed Question:

A Reactor trip and safety injection have occurred, and initial conditions are as follows:

- The "B" Pzr PORV is deenergized closed.
- The crew has entered E-1, *Loss or Reactor or Secondary Coolant*.
- RCS pressure is 2350 psia, with pressure being controlled by the "A" Pzr PORV (3RCS-PCV455A).
- SG pressures are 1100 psig and stable.
- The crew is about to perform E-1, step 9, "Check RCS and SG Pressures".

The RO and BOP report the following:

- RCS pressure is 2200 psia and dropping rapidly.
- SG pressures are 1100 psig and stable
- PORV 3RCS*PCV455A indicates OPEN.

What actions are required to be taken by the crew?

a) CLOSE the "A" PORV, and if the PORV will not close, close its Block valve. If pressure starts to recover, the crew will transition to ES-1.2, *Post-LOCA Cooldown and Depressurization*.
b) CLOSE the "A" PORV, and if the PORV will not close, close its block valve. If pressure starts to recover, the crew will transition to ES-1.1, *SI Termination*.
c) Do NOT close the "A" PORV. The crew will transition to ES-1.2, *Post-LOCA Cooldown and Depressurization*.
d) Do NOT close the "A" PORV. The crew will transition to ES-1.1, *SI Termination*.

Proposed Answer: B

Explanation:

This question is considered SRO level, since it requires the applicant to assess plant conditions, and interpret a procedure step with PORVs in an abnormal lineup and the applicable step having already been moved past. The question also requires the applicant to determine which ES procedure the crew will ultimately transition to, even though the required transition step has already been passed.

"C" and "D" are wrong, since the crew is required to close the B" PORV, and if it doesn't close, close and the "B" PORV Block Valve and keep it closed. Based on the CAUTION prior to E-1, step 5, the action to close a stuck open PORV and its associated Block Valve if required is a continuous action step. "C" and "D" are plausible, since E-1, step 5.b has the crew check at least one Block Valve open, which normally is desired, and if neither are open, the crew would normally open at least one block valve. But the RNO has the crew keep the block valves closed if they are unavailable or are being used to isolate a stuck open PORV. Also, the other PORV is deenergized closed, and by closing this PORV, no more PORV protection is available. It is also generally desired to minimize lifting of the Pzr Safety Valves, and in certain situations such as FR-H.1, PORVs are deliberately opened to provide a RCS heat removal path.

"B" is correct, and "A" wrong, since after the Block Valve is closed, RCS injection and leakage are restored to the conditions that existed before the PORV failed open, which means that RCS pressure will again start increasing with SG pressures stable. When the crew reaches E-1, step 9 with SG pressure stable and RCS pressure increasing, the crew will be directed to return to E-1, step 1. This allows the crew to transition to ES-1.1, *SI Termination* when they reach E-1, step 6. If the crew were to proceed ahead in E-1 to step 10, they will be directed to ES-1.2, *Post LOCA Cooldown and Depressurization*, where they would meet unnecessarily restrictive SI termination criteria, complicating the event. "A" is plausible, since with SG pressures stable and RCS pressure decreasing, E-1, the crew would proceed to E-1, step 10, and at step 12, transition to ES-1.2 with RCS pressure above 300 psia.

Technical Reference(s): E-1 (Rev. 27), steps 5-10
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO, STA) Given a set of plant conditions, determine the required actions to be taken per E-1

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 77	Tier #		1
K/A Statement: Small Break LOCA: Knowledge of specific bases for EOPs and AOPs	Group #		1
	K/A #	EPE 9.G2.4.18	
Proposed Question:	Importance Rating		4.0

The plant trips due to a small break LOCA, resulting in the following sequence of events:

1. The crew enters ES-1.2, *Post LOCA Cooldown and Depressurization*.
2. The crew reaches ES-1.2, step 22, "Check Isolating SI Accumulators", and current conditions are as follows:

- RCS Pressure: 325 psia and slowly decreasing
- Core Exit Thermocouples: 396°F and slowly decreasing
- RCS hot leg temperatures: 393°F and slowly decreasing
- Pressurizer Level: 18% and steady

Per ES-1.2, step 22, is the crew required to isolate the SI Accumulators? Why or why not?

a) NO, since RCS temperature and pressure indicate that the Accumulators' liquid contents are still required to mitigate this event.
b) NO, since RCS hot leg temperature is above the temperature where Accumulator isolation is allowed.
c) YES, since RCS temperature and pressure indicate that the Accumulators' liquid contents are not required to mitigate this event, and injection will hinder reaching conditions where RHR Pumps can be placed in service.
d) YES, since RCS hot leg temperature has dropped to less than the point corresponding to Accumulator pressure after water discharge. Isolation limits Accumulator nitrogen injection into the RCS.

Proposed Answer: D

Explanation:
This question is considered SRO level, since it requires the applicant to apply procedure basis information in a Supplemental Emergency (ES) procedure to make a procedural decision that goes beyond system knowledge and beyond overall procedure strategy.
At 396°F, saturation pressure is 236 psia. At 428°F (32° hotter), saturation pressure is 336 psia. Actual pressure is less than 336 psia, so subcooling is below 32°F.
"A" and "B" are wrong, since per step 22 RNO, with hot leg temperatures below 440°F, the crew will isolate SI accumulators even without adequate subcooling. "A" is plausible, since subcooling is <32°F, and this is the criterion that normally requires the crew to leave the Accumulator Isolation Valves open. Also, the basis listed in distractor "A" is the reason Accumulators are normally left unisolated with less than 32°F subcooling. "B" is plausible, since subcooling is <32°F, and this is the criterion that normally requires the crew to leave the Accumulator Isolation Valves open. Also, there is a Tech Spec limit for when SI Accumulators can be isolated during an RCS cooldown.
"C" is wrong, and "D" correct, since the reason the crew will isolate accumulators with inadequate subcooling and < 440°F is because RCS pressure has dropped to the accumulator pressure expected after accumulator water has injected. No more water is available, and nitrogen will inject. Nitrogen injection will either produce a "hard" bubble in the PZR hindering RCS depressurization, or gas binding in the SG U-Tubes, limiting heat transfer from the primary to the SGs. "C" is plausible, since the crew is required to isolate accumulators, and this is the basis for isolating accumulators if subcooling is >32°F.

Technical Reference(s):	ES-1.2 (Rev. 22), Step 22
(Attach if not previously provided,	BKG EOP 35 ES-1.2 (Rev. 22) Basis information for step 22
including version/revision number.)	Tech Spec LCO 3.5.1 (Amendment 258), Applicability
	Steam Tables

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

Learning Objective: Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 ES-1.2.

Question Source:	New
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.5 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 78	Tier #		1
K/A Statement: Pzr Pressure Control System Malfunction: Ability to determine and/or interpret the effects of RCS pressure changes on key components in the plant	Group #		1
	K/A #	APE27.AA2.03	
	Importance Rating		3.4

Proposed Question:

The plant is initially at 75% power, when a load reject occurs, resulting in the following sequence of events:

1. RCS pressure initially increases to 2340 psia.
2. The Pzr Pressure Control System responds, and RCS pressure decreases to and stabilizes at 2,230 psia.
3. EEQ data shows temperature downstream of the “A” PZR PORV (3RCS*PCV455A) has increased by 10°F.
4. For diagnosis, the crew attempts to CLOSE the “A” PORV Block Valve (3RCS*MV8000A), but its fuse blows as soon as the block valve starts to stroke.
5. Engineering reports “A” PORV is leaking by, but leakage is considered “NOT excessive”.
6. Electrical Maintenance determines that the Block Valve Breaker needs to be repaired, and they estimate the repairs will take about 24 hours.

Complete the following statement.

When Pzr pressure initially stabilized at 2,230 psia, the Pzr Backup Heaters __(1)___energized; and based on the estimated time of repair, Tech Spec ACTION requires the crew to place / keep the “A” PORV Control Switch in the __(2)___position. **(Reference provided)**

	(1)	(2)
a)	WERE NOT	AUTO
b)	WERE NOT	CLOSE
c)	WERE	AUTO
d)	WERE	CLOSE

Proposed Answer: B

Explanation:
This question is considered SRO level, since it requires the applicant to assess plant conditions and determine which Tech Spec ACTIONs are required. It also requires the applicant to apply surveillance requirements to determine if a PORV with seat leakage is considered OPERABLE.
LCO 3.4.4, PORV Action "a" does not need to be entered, since seat leakage is NOT excessive.
Action "b" does not need to be entered, since the PORV meets all surveillance requirements, even with the seat leakage. Action "c" does not need to be entered, since only one PORV block valve is inoperable. Action "d" needs to be entered, since the "A" Block Valve is not OPERABLE.
“A” and “C” are wrong, since per Action “d” with one or both block valve(s) INOPERABLE, within 1 hour restore the block valve(s) to OPERABLE status (will not occur), or place its associated PORV(s) control switch to "CLOSE". The requirement to restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable does not apply, since the "B" block valve is OPERABLE. After taking the PORV control switch to CLOSE, the crew is required to restore any remaining inoperable block valve to OPERABLE status within 72 hours (it is projected be repaired within this time); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. “A” and “C” are plausible, since per Action “b”, if the PORV were considered INOPERABLE, no Action with the Block Valve would be required. Also, with PORV leakage deemed NOT excessive, the PORV Control Switch would be kept in AUTO if the Block Valve were OPERABLE.

“B” is correct, and “D” wrong, since the backup heaters come on when pressure is 25 psi less than the normal setpoint pressure of 2250 psia, and pressure was above 2225 psia. "D" is plausible; since backup heaters do energize on low Pzr pressure, and pressure is low. Also, pressure is below the setpoint where the Control Heaters are fully energized (2235 psia), and Backup Heaters deenergize if they are on (2233 psia).

Technical Reference(s):	Tech Spec LCO 3.4.4 (Amendment 229)
(Attach if not previously provided,	Tech Spec Surveillance Requirement 4.4.4.1 (Amendment 264)
including version/revision number.)	Functional Sheet 11 (Rev. H) and 12 (Rev. F)
	Training Lesson Plan PPL010C (R5C3) PowerPoint Slide 29

Proposed references to be provided to applicants during examination: **Tech Spec 3/4.4.4 (page 3/4 4-12)**

Learning Objective: (SRO, STA) Given a plant condition or equipment malfunction, use provided reference material to… Evaluate Technical Specification Applicability and determine required Actions

Question Source:	New
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	41.7 and 43.2
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 79	Tier #		1
K/A Statement: Loss of Offsite Power: Ability to analyze the effect of maintenance activities, such as degraded power sources on the status of limiting conditions for operation	Group #		1
	K/A #	APE 56.G2.2.36	
	Importance Rating		4.2

Proposed Question:

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

- The crew is preparing to remove the "B" Emergency Diesel Generator from service for scheduled maintenance.
- The crew plans on extending the allowed outage time (AOT) from 72 hours to 14 days to allow adequate time to complete the scheduled maintenance.

In accordance with Tech Spec LCO 3.8.1.1 and the bases section of Technical Specifications, what condition would be acceptable when relying on this extended AOT?

a) The Turbine Driven AFW Pump is out of service for routine maintenance.
b) The SBO Diesel was last verified to be available by test performance 15 days ago.
c) The "B" Charging Pump and CCE Pump are in operation, and the "A" Charging Pump and CCE pump are available, but the swing Charging Pump is tagged OOS.
d) Elective maintenance that could challenge offsite power availability has been scheduled for the switchyard in 3 days. The maintenance will be closely monitored and controlled.

Proposed Answer: B

Explanation:
This question is considered SRO level, since it requires the applicant to apply Tech Spec basis information in determining whether equipment is allowed to be removed from service. This question is considered a KA match since the bases for these requirements is to "provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems…."
"B" is correct, since the SBO diesel must be verified available by test performance within 30 days prior to allowing the EDG to be INOP for greater than 72 hours.
"A" is wrong, since "when one diesel generator is inoperable, there is an additional ACTION requirement… to verify… the steam-driven auxiliary feedwater pump is OPERABLE…" "A" is plausible since the TDAFW Pump is not train specific, and train-specific Motor Driven AFW Pumps are available.
"C" is wrong, since the swing charging pump is required to be available. "C" is plausible, since the charging pump aligned to the operable EDG is available.
"D" is wrong, since elective maintenance is not allowed in the switchyard. "D" is plausible since the proposed maintenance will be closely monitored and controlled. This is a requirement for activity in the switchyard during an extended EDG outage.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
LCO 3.8.1.1.b (Amendment No 229), ACTION b.3
Tech Spec Bases for LCO 3.8.1.1.a (LBDCR 04-MP3-015, February 24, 2005), page B 3/4 8-1a
Tech Spec Bases for LCO 3.8.1.1.a (LBDCR 20-MP3-012, February 18, 2021), page B 3/4 8-1b

Proposed references to be provided to applicants during examination: None

Learning Objective: 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 #403131
Question History: Last NRC Exam Millstone 3 2007 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.43.2
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 80	Tier #		1
K/A Statement: Loss of Nuclear Service Water: Ability to determine and/or interpret implementation of TS requirements	Group #		1
	K/A #	APE 62.AA2.07	
	Importance Rating		4.0

Proposed Question:

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

1. The Outside Rounds PEO reports a significant lube oil leak exists on the “A” Service Water Pump.
2. The crew starts the “C” SWP Pump, and stops the “A” SWP Pump.
3. The crew places the “A” SWP Pump in Pull-To-Lock.

Does the loss of the "A" Service Water Pump affect the OPERABILITY of the "A" Emergency Diesel Generator? Why or why not?

a) The "A" EDG IS still OPERABLE, since there is still an OPERABLE Service Water Pump available on the "A" Train.
b) The "A" EDG IS still OPERABLE, since the crew has already logged into the Service Water Tech Spec LCO.
c) The "A" EDG is NOT OPERABLE since both Service Water pumps are required for the affected SWP Train to be OPERABLE.
d) The "A" EDG is NOT OPERABLE since both Service Water pumps are required for the affected EDG to be OPERABLE.

Proposed Answer: A

Explanation:
This question is considered SRO level, since it requires the applicant to consider both Tech Spec and TRM bases information to determine the effect of a loss of Safety Related equipment on another System.
“A” is correct, and “C” and “D” wrong, since per LCO 3.8.1.1, the EDG only requires an OPERABLE SWP "loop", and per LCO 3.7.4, only one SWP pump is required to provide core and CTMT cooling during a DBA LOCA with a single failure, so the SWP train is considered OPERABLE, and since Service Water is still considered OPERABLE, EDG OPERABILITY is not affected.
“C” is plausible, since for some Safety Related Systems, if a Safety Related Pump is lost, a LCO ACTION is required to be entered.
“D” is plausible, since this is true for Service Water per the TRM, where it is assumed one SWP pump is supplying core and CTMT heat loads, while the other Pump is required to supply the Spent Fuel Pool.
"B" is wrong, since if a SWP Train becomes INOPERABLE, its EDG is also declared INOPERABLE, since its LCO ACTION time is more restrictive than the SWP LCO. The SWP LCO 3.7.4 ACTION time with only one SWP Loop OPERABLE is to restore the INOPERABLE loop to OPERABLE within 72 hours or be in HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours. EDG LCO 3.8.1.1 ACTION b requires the crew to perform Offsite Circuit Surveillance within 1 hour and at least once per 8 hours thereafter, and demonstrate a common mode failure does not exist on the OPERABLE EDG with 24 hours, or perform a Surveillance on the OPERABLE EDG within 24 hours, along with other LCO ACTIONS. These EDG LCO ACTION times are more restrictive than the SWP Tech Spec LCO ACTION Time of 72 hours. “B” is plausible, since the general practice at Millstone 3 is not to cascade Tech Spec ACTIONS, unless a more restrictive time limit applies.

Technical Reference(s):	Tech Spec LCO 3.7.4 (Amendment 258)
(Attach if not previously provided,	Tech Spec 3.7.4 Bases (LBDCR 3-22-02)
including version/revision number.)	TRM 3.7.4 (LBDCR 07-MP3-018)
	TRM 3.7.4 Bases (LBDCR 07-MP3-018)
	Tech Spec LCO 3.8.1.1 (Amendment No. 285)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a plant condition or equipment malfunction (related to the EDGs), use provided reference material to… Evaluate Technical Specification applicability…

Question Source:	Bank #405293	
Question History:	Last NRC Exam	Millstone 3 2013 NRC Exam
Question Cognitive Level:	Comprehension or Analysis	
10 CFR Part 55 Content:	55.43.2	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 81	Tier #	1	1
K/A Statement: LOCA Outside Containment:	Group #	1	1
Ability to interpret and execute procedure steps.	K/A #	W E04.G2.1.20	
Proposed Question:	Importance Rating		4.6

With the plant initially at 100% power, Safety Injection actuates due to a LOCA outside Containment, resulting in the following sequence of events:

1. Over the next 10 minutes, RCS pressure increases to and cycles on the Pzr PORVs at 2350 psia.
2. The crew enters ECA-1.2, *LOCA Outside Containment*.
3. The final valve the crew closes while attempting to isolate the break is the "A" RHR Pump Cold Leg Injection Valve (3SIL*MV8809A).
4. After 3SIL*MV8809A closes, the RO reports that RCS pressure is still cycling at 2350 psia.
5. The STA observing the Real-Time trend reports that the Pzr PORVs are cycling at the same rate as before.

To which procedure is the crew required to transition from ECA-1.2?

a) E-1, *Loss of Reactor or Secondary Coolant*.
b) ES-1.1, *SI Termination*.
c) ES-1.2, *Post LOCA Cooldown and Depressurization*.
d) ECA-1.1, *Loss of Emergency Coolant Recirculation*.

Proposed Answer: D

Explanation:
This question is considered SRO, since it requires the applicant to assess plant conditions and determine which procedure to transition to from an Emergency Contingency Action (ECA) procedure.
The main method for determining leak isolation status is to monitor RCS pressure. But in this case, the LOCA is small enough that RCS pressure is already cycling on the PORVs, so pressure change based on leak isolation status, so the NOTE prior to step 1 must be applied while pressure is cycling on the PORVs.
"D" is correct and "A" wrong, since it can be determined that the break has not been isolated based on steady PORV cycling rate. "A" is plausible since step 5 directs transition to E-1 if the leak is isolated, and a potential injection path and break path have been isolated.
"B" and "C" are wrong, since with the break isolated, the crew is required to transition to E-1 first, and then to ES-1.1 or ES-1.2, depending on RCS conditions. "B" and "C" are plausible, since the RCS is already cycling on the PORVs, so break size is small, and for small breaks or isolated breaks, the crew will normally end up in ES-1.1 or ES-1.2.

Technical Reference(s): ECA-1.2 (Rev. 10), step 4, including Note, and steps 5 and 6
(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 ECA-1.2
Question Source: Modified Bank #407696 (Parent question attached below)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.3, 41.5, and 43.5

Comments:
This question is considered “Modified” since the PORV cycling rate has been changed in the stem, making a previous distractor the new correct answer. Also, one of the distractors has been changed from ES-1.3 to ES-1.2 (more plausible with a small break LOCA in progress).

Original Bank Question #407696

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

1. A LOCA outside Containment occurs, resulting in a Reactor trip and safety injection.
2. Over the next 10 minutes, RCS pressure increases to and cycles at 2350 psia, with PZR PORVs cycling.
3. The crew is responding using ECA-1.2, *LOCA Outside Containment*.
4. RWST level is 900,000 gallons and slowly decreasing.
5. Pressurizer level is 65% and increasing.
6. The final valve the crew closes while attempting to isolate the break is the “A” RHR Pump Cold Leg Injection Valve (3SIL*MV8809A).
7. After 3SIL*MV8809A closes, the RO reports that RCS pressure is still cycling at 2350 psia.
8. The STA observing the Real-Time trend reports that the Pzr PORVs are now cycling at a significantly faster rate.

To which procedure is the crew required to transition from ECA-1.2?

a) E-1, *Loss of Reactor or Secondary Coolant*.
b) ES-1.1, *SI Termination*.
c) ES-1.3, *Transfer to Cold Leg Recirculation*.
d) ECA-1.1, *Loss of Emergency Coolant Recirculation*.

A

Examination Outline Cross-reference:	Level	RO	SRO
Question # 82	Tier #		1
K/A Statement: Control Room Evacuation: Ability to determine and/or interpret saturation margin	Group #		2
	K/A #	APE 68.AA2.09	
Proposed Question:	Importance Rating		3.8

With the plant initially at 100% power, toxic gas is released inside the Control Room, resulting in the following sequence of events:

1. The crew completes all actions in EOP 3503, *Shutdown Outside Control Room*.
2. RCS Subcooling slowly drifts down to 28°F.
3. The US directs the required actions to be taken per EOP 3503 to restore RCS subcooling.
4. The crew completes all other actions in EOP 3503 and transitions to EOP 3504, *Cooldown Outside the Control Room*.
5. The operator at the switchgear reports the following:
   - The "A" CRDM Cooling Fan is running.
   - The "B" and "C" CRDM Cooling Fans are off, and can NOT be started.

Complete the following statement.

The crew was directed by EOP 3503 to __(1)__ due to RCS subcooling having decreased to 28°F; and during the upcoming RCS cooldown per EOP 3504, the crew will be required to __(2)__ to compensate for having only one CRDM Cooling Fan running.

a) (1) actuate Safety Injection
   (2) establish reactor vessel head vent excess letdown
b) (1) actuate Safety Injection
   (2) raise minimum required RCS subcooling from 82°F to 132°F
c) (1) increase the rate of dumping steam
   (2) establish reactor vessel head vent excess letdown
d) (1) increase the rate of dumping steam
   (2) raise minimum required RCS subcooling from 82°F to 132°F

Proposed Answer: C

Explanation:
This question is considered SRO level since it requires the applicant to assess plant conditions and determine specific EOP actions to be taken that go beyond system knowledge and beyond the overall strategy of the procedure. This is also a SRO/STA specific objective at Millstone 3.
"A" and "B" are wrong, since with subcooling <32° in EOP 3503, the crew will be directed to increase the rate of dumping steam, as opposed to initiating SIS. "A" and "B" are plausible, since in most places in the EOP network, <32°F subcooling requires the crew to either actuate SIS, or reinitiate SIS.
"C" is correct, since with only one CRDM Cooling Fan running in EOP 3504, step, 6, the crew will be directed to align a head vent path to the PRT to provide additional vessel head cooling to minimize the potential for drawing a bubble in the vessel head.
"D" is wrong, since, although not stated in the stem of the question, the crew will have stopped the RCPs per EOP 3503, step 14 (since RCP support conditions cannot be adequately monitored at the ASP); and a natural circulation cooldown will require the crew to cooldown the plant at 50°F per hour and maintain minimum subcooling >82°F to prevent a void from forming in the Reactor vessel head. "D" is plausible, since minimum subcooling of 132°F would be required with only one CRDM Cooling Fan running and no head vent path aligned during a natural circulation cooldown per ES-0.2, *Natural Circulation Cooldown*.

Technical Reference(s): EOP 3503 (Rev. 18), Note prior to step 2, and steps 14 and 28, inc. RNO
(Attach if not previously provided, EOP 3504 (Rev. 9-2), steps 4, 6, and 11
including version/revision number.) ES-0.2 (Rev. 19-1), step 11

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO) Given a set of plant conditions, determine the required actions to be taken per EOP 3503

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.14 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 83	Tier #		1
K/A Statement: Natural Circulation Operations: Knowledge of EOP exit conditions (e.g. emergency condition no longer exists or SAG entry is required)	Group #		2
	K/A #	W E09.G2.4.51	
	Importance Rating		4.0

Proposed Question:

The crew has entered ES-0.2, *Natural Circulation Cooldown*, and initial conditions are as follows:

- An RCS cooldown is in progress per ES-0.2, step 5.
- The crew has just initiated the RCS depressurization per ES-0.2, step 11.
- RCP seal injection has been maintained throughout the event.

The SM informs the US that conditions have been established to allow starting the "B" RCP.

Per ES-0.2, what action is required to be taken by the crew?

a) Using GA-6, *Starting Reactor Coolant Pump*, START the "B" RCP and transition to OP 3208, *Plant Cooldown*.
b) Using GA-6, *Starting Reactor Coolant Pump*, START the "B" RCP and complete ES-0.2 before transitioning to OP 3208.
c) DO NOT start the "B" RCP until COLD SHUTDOWN conditions are reached.
d) DO NOT start the "B" RCP until the ADTS performs a status evaluation of the plant.

Proposed Answer: A

Explanation:
This question is considered SRO level, since it requires the applicant to assess plant conditions, and determine that a previous step in ES-0.2 applies that directs the use of a GA procedure. It also requires the applicant to determine if conditions will be met to start an RCP per the GA, and also decide which procedure the crew will transition to from this ES procedure.
"A" is correct, since ES-0.2, step 1 still applies, seal cooling has not been lost, and based on present conditions, nothing will prevent the crew from starting the RCP per the first four steps of GA-6.
"B" is wrong, since ES-0.2, step 1 has the crew continue with normal plant evolutions using applicable plant procedures after an RCP is started. "B" is plausible, since ES-0.2 cooldown and depressurization steps are already in progress, and the last step of ES-0.2 would transition the crew to OP 3208 if the crew remained in ES-0.2
"C" is wrong, since ES-0.2, step 1 is a continuous action step, forced flow allows a faster cooldown rate with better pressure control and less potential for head voiding than with a natural circulation cooldown. Also, some EOPs (e.g., ECA-0.0 and ECA-2.1) require the crew to complete groups of steps before taking other actions or transitions.
"D" is wrong, since this is required only if seal injection has been previously lost (ES-0.2, step 1 Caution). "D" is plausible, since a status evaluation would be required if seal injection had previously been lost, and RCP start requirements are different based on which EOP is in progress (e.g., ECA-0.1 and FR-C.1).

Technical Reference(s): ES-0.2 (Rev. 19-1), Cautions and Notes prior to step 1
(Attach if not previously provided, ES-0.2 (Rev. 19-1), steps 1-12, and 17- 20
including version/revision number.) GA-6 (Rev. 4), steps 1-4

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss conditions which require transition to other procedures from EOP 35 ES-0.2, Natural Circulation Cooldown

Question Source: Bank #408719
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 # 84	Tier #		1
K/A Statement: Rediagnosis: Ability to determine and/or interpret which procedure or procedure set should be transitioned to	Group #		2
	K/A #	W E01.EA2.09	
	Importance Rating		4.0

Proposed Question:

A SG Tube Rupture occurs on the "C" SG, resulting in the following initial sequence of events:

1. The crew enters E-3, *Steam Generator Tube Rupture*.
2. The crew has just completed the cooldown and depressurization of the RCS per E-3.
3. The RO reports ruptured SG pressure has lowered to 500 psig.
4. The crew isolates the "C" SG per E-2, *Faulted Steam Generator Isolation*.
5. The US has difficulty navigating the EOP flowpath, and chooses to enter ES-0.0, *Rediagnosis*.

Current conditions are as follows:

- RCS pressure: 1050 psia and slowly lowering
- Pressurizer level: 15% and slowly lowering
- "C" SG pressure: 480 psig and slowly lowering
- "C" SG NR level: 0%
- RWST Level: 1,100,000 gallons and slowly lowering

The US desires to transition from ES-0.0 to the procedure that meets the following criteria:

- The transition is procedurally directed by ES-0.0.
- The new procedure will provide the most efficient method of addressing the accident in progress.

What procedure transition from ES-0.0 will the US direct?

a) ES-3.1, *Post SGTR Cooldown using Backfill*
b) ES-3.2, *Post SGTR Cooldown using Blowdown*
c) ECA-3.1, *SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired*
d) ECA-3.2, *SGTR with Loss of Reactor Coolant - Saturated Recovery Desired*

Proposed Answer: C

Explanation:
This question is considered SRO, since it requires the applicant to assess plant conditions, and determine which Emergency Contingency Action (ECA) Procedure is required to be transitioned to.
The normal flowpath through E-3 is to complete the cooldown and depressurization steps (this has been completed), and then terminate SIS and transition to a supplemental (ES) procedure (E-3, step 43).
"A" and "B" are wrong, since ES-3.1 and 3.2 are not procedurally directed transitions from ES-0.0. "A" and "B" are plausible, since these are the normal transitions from E-3, and the RCS cooldown and depressurization portions of E-3 had been completed prior to entering ES-0.0, so most of E-3 has already been completed.
"C" is correct, since this a procedure transition directed by ES-0.0, and with ruptured SG pressure less than 530 psig, E-3 would direct the crew to transition to ECA-3.1, *SGTR with Loss of Reactor Coolant - Subcooled Recovery Desired* (E-3, step 5). This is directed since the ruptured SG is faulted, so RCS break flow cannot be stopped until the RCS is completely depressurized.
"D" is wrong, since there are two cases specified in ECA-3.1, step 12 where the crew would to ECA-3.2, *SGTR with Loss of Reactor Coolant - Saturated Recovery Desired*. One is if RWST inventory is less than 920,000 gallons and CTMT sump level is not increasing at an adequate rate to support cold leg recirculation if required. The second is if ruptured SG overfill is becoming a significant concern as evidenced by ruptured SG NR level having increased to 87%, which would significantly increase the radiation release to the environment. Neither of these criteria are met, since RWST level is 980,000 gallons, and ruptured SG NR level is 0%. If the crew did transition to ECA-3.2, step 1 would transition the crew to ECA-3.1, and this procedure loop would not be

expeditious way to address the event in progress. “D” is plausible, since RWST level is lowering, and ECA-3.2 is used to rapidly progress to Cold Shutdown with a Tube Rupture in progress.

Technical Reference(s): ES-0.0 (Rev. 7), Steps 3 and 4
(Attach if not previously provided, E-3 (Rev. 29), steps 5, 18, 19, 24, and Foldout Page item 1
including version/revision number.) ECA-3.1 (Rev. 23), step 12
ECA-3.2 (Rev. 23), step 1.a
OP 3272 (Rev. 14), Section 3.2, page 9 of 66

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss conditions which require transition to other procedures from EOP 35 ES-0.0

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 # 85	Tier #		1
K/A Statement: Degraded Core Cooling: Ability to interpret control room indications to verify the status and operation of a system and understand how operator actions and directives affect plant and system conditions	Group #		2
	K/A #	W E06.G2.2.44	
	Importance Rating		4.4

Proposed Question:

The Reactor has tripped, and initial conditions are as follows:

- Core Exit Thermocouples are 744°F.
- The crew has transitioned to the appropriate Functional Restoration Procedure.
- The crew is currently depressurizing all intact SGs to the target pressure of 190 psig.

The INTEGRITY critical safety function status tree turns RED.

What actions are required to be taken by the crew?

a) Transition from the current step in the current procedure to FR-P.1, *Response to Imminent Pressurized Thermal Shock.*
b) Complete the SG depressurization per the current procedure, then transition to FR-P.1, *Response to Imminent Pressurized Thermal Shock*.
c) Perform appropriate actions of FR-P.1, *Response to Imminent Pressurized Thermal Shock* in parallel with the actions being performed in the current procedure.
d) Complete the current procedure entirely then transition to FR-P.1, *Response to Imminent Pressurized Thermal Shock.*

Proposed Answer: D

Explanation:
This question is considered SRO level since it requires the applicant to assess plant conditions to determine which Functional Restoration Procedure (FRP) currently applies, and identify a specific condition that requires actions that are different from the standard FRP implementation rules of usage.
"D" is correct, and "A", "B", and "C" wrong, since the operators are required to remain in C.2 to restore adequate core cooling. It is expected that during the SG depressurization, the integrity tree may turn RED, since accumulators will inject. Core cooling requires the accumulators to inject, and FR-P.1 actions may hinder core cooling. FR-C.2 need to be completed prior to addressing P.1 in this case.
“A” and “B” are plausible, since per EOP rules of usage, FR-P.1 (Red Path) is a higher priority than FR-C.2 (Orange Path).
“C” is plausible, since this is how EOP rules of usage would be applied to an EOP and an AOP that both apply to an event in progress.

Technical Reference(s): OP 3272 (Rev. 14), Section 3.9
(Attach if not previously provided, OP 3272 (Rev. 14), Attachment 3
including version/revision number.) FR-C.2 (Rev. 17), CAUTION prior to step 13

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO, STA) Given a set of plant conditions, determine the required actions to be taken per FR-C.2

Question Source: Bank #408135
Question History: Last NRC Exam Millstone 3 2002 NRC
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5 and 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 86	Tier #		2
K/A Statement: Residual Heat Removal System: Ability to perform general and/or normal operating procedures during any plant condition	Group #		1
	K/A #	005.G2.1.23	
	Importance Rating		4.4

Proposed Question:

A plant cooldown is in progress in accordance with OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- Both trains of RHR are in service in the Cooldown Mode.
- Pressurizer level is stable at 55%, being maintained by 3RCS-LK459 “PZR LVL” in MANUAL.
- RCS cold leg temperatures are 250°F and decreasing.
- RCS pressure is 350 psia and stable.
- PZR temperature and surge line temperature are both stable at 430°F.
- All Pressurizer heaters are energized.

The RO reports Pressurizer surge line temperature indicates 420°F and lowering.

What adverse plant condition exists; and per OP 3208, what is one action the US can direct due to this condition?

a) Spray flow has initiated with excessive delta T across the Pressurizer Spray nozzle. The US will notify Engineering as part of tracking the cumulative thermal stress on the Spray Nozzle.
b) Spray flow has initiated with excessive delta T across the Pressurizer Spray nozzle. The US will direct the RO to deenergize Pressurizer Heaters to restore delta T to within limits within 30 minutes.
c) A Pressurizer insurge is in progress. The US will direct the RO to adjust 3RCS-LK459 “PZR LVL” to decrease charging flow.
d) A Pressurizer insurge is in progress. The US will direct the RO to adjust 3CHS-PK131 “L/D PRES CNTL” to decrease letdown flow.

Proposed Answer: C

Explanation:
This question is considered SRO level, since it requires the applicant to assess plant conditions and take specific procedure actions that go beyond overall system knowledge of procedure strategy. This requirement is specified by an SRO-specific objective at Millstone 3. It also requires the applicant to check plant conditions against TRM requirements. As the plant is cooling down, the RCS water contracts, requiring additional RCS inventory to be added to maintain a stable Pressurizer level. At this specific point in OP 3208, the PZR Level Control System is being used to maintain PZR level constant, requiring net charging flow to be greater than net letdown flow. Two temperature related concerns at this point in OP 3208. One is a TRM requirement to maintain a maximum temperature difference between the RCS and the PZR less than 320°F to minimize thermal stress on the PZR spray nozzle. There also is a requirement to minimize the thermal stresses on the PZR surge line during the cooldown. To ensure TRM requirements are met, OP 3208 sets up a condition where spray flow is adding water to the PZR at a rate greater than the net charging rate is adding water to the RCS. This establishes a continuous PZR outsurge while adding water to the RCS, preventing a PZR insurge and the associated thermal transient on the surge line. If net charging flow increases above spray flow, an insurge occurs, as evidenced by the surge line temperature drop.

"A" is wrong, since the initial RCS to PZR temperature difference was 180°F (430°-250°F), so these actions are not required. "A" is plausible, since these actions would be required if the 200°F administrative limit was exceeded.

"B" is wrong, since the initial RCS to PZR temperature difference was 180°F (430°-250°F), and temperature has only decreased by 10°F, so the 320°F TRM spray water temperature differential limit has not been exceeded. "B" is plausible, since a TRM maximum spray water temperature differential limit exists, and the cold leg cooldown has made the differential temperature increase, and these actions are the actions required if the TRM limit was exceeded.

"C" is correct, since surge line decreasing below PZR temperature indicates that an insurge is occurring, and the US is required to either increase letdown flow or decrease charging flow to restore outsurge conditions.

"D" is wrong, since decreasing letdown flow is not required. "D" is plausible, since adjusting letdown flow would correct the situation, but it would need to be increased.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
TRM 3.4.9.2.c (LBDCR 07-MP3-018)
OP 3208 (Rev. 38), Precautions 3.5.2, 3.5.3, and 3.5.4
OP 3208 (Rev. 38), steps 4.3.33 and 4.3.34
OP 3208 basis document (Rev. 38) for step 4.3.34

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO, STA) Given a set of plant conditions, determine the required actions to be taken per OP 3208

Question Source: Bank #406639

Question History: Last NRC Exam Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5, 43.2, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 87	Tier #		2
K/A Statement: Component Cooling Water: Ability to determine operability or availability of safety-related equipment (SRO Only)	Group #		1
	K/A #	008.G2.2.37	
	Importance Rating		4.6

Proposed Question:

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

- The "A" and "B" RPCCW Pumps are running.
- The "C" RPCCW Pump is in standby, aligned to the "B" Train per OP 3330A, *Reactor Plant Component Cooling Water.*

The following sequence of events occurs:

1. The "A" RPCCW Pump trips.
2. The crew enters AOP 3561, *Loss of Reactor Plant Component Cooling Water*.

Complete the following statement about where in AOP 3561 the crew will be directed to align and start the "C" RPCCW Pump; and the action required with a 4 KV breaker to align the "C" RPCCW Pump to the "A" Train.

Per AOP 3561, ___(1)____, the crew will mechanically and electrically align the "C" Pump to the "A" Train, which includes racking down the ___(2)____, and racking it up into the "C" RPCCW Pump breaker cubicle on Bus 34C.

a) (1) Step 1, "Check RPCCW System Alignment"
   (2) "A" RPCCW Pump breaker from its cubicle on Bus 34C
b) (1) Step 1, "Check RPCCW System Alignment"
   (2) "C" RPCCW Pump breaker from its cubicle on Bus 34D
c) (1) Attachment "A", "Recover From Loss Of RPCCW Pump"
   (2) "A" RPCCW Pump breaker from its cubicle on Bus 34C
d) (1) Attachment "A", "Recover From Loss Of RPCCW Pump"
   (2) "C" RPCCW Pump breaker from its cubicle on Bus 34D

Proposed Answer: D

Explanation: This question is considered SRO level, since it requires the applicant to determine whether to use the body of an AOP or an Attachment in that AOP to align a standby pump for service.
"A" and "B" are wrong, since the crew will be required to use Attachment "A" to align the "C" (swing) RPCCW Pump to the "B" Train. "A" and "B" are plausible, since step 1 of the procedure would be used to start the "C" (swing) pump if it were already aligned to the "A" Train.
"C" is wrong, and "D" correct, since per Attachment A, the crew will be directed to use OP 3330A to rack down the "C" RPCCW Pump breaker from its "B" Train cubicle on Bus 34D and rack it up into its "A" Train cubicle on bus 34C. "C" is plausible, since this is how the "C" (swing) Charging Pump breaker would have to be aligned, since it does not have its own dedicated breaker, but uses the breaker from the Charging Pump dedicated to that train.

Technical Reference(s): AOP 3561 (Rev. 21), step 1, especially steps 1.c-e
(Attach if not previously provided, AOP 3561 (Rev. 21), Attachment A, especially steps A.6.a-c
including version/revision number.) OP 3330A (Rev. 29), Section 4.9, steps 4.9.11, 4.9.23, and 4.9.24

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO) Given a set of plant conditions, determine the required actions to be taken per AOP 3561

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 # 88	Tier #		2
K/A Statement: Emergency Safeguards Actuation: Ability to predict the impacts of and use procedures to correct, control, or mitigate the consequences of a loss of DC control power on the ESFAS System	Group #		1
	K/A #	013.A2.05	
	Importance Rating		3.8

Proposed Question:

Initial conditions:

- The plant is at 100% power.
- A DC control power fuse supplying the “A” SIH Pump breaker has blown.

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

1. Offsite power is lost on the Reactor trip.
2. The “B” EDG fails to start.
3. The SBO Diesel fails to start.
4. The RO reports the “A” SIH Pump is NOT running, and cannot be started at MB2.
5. The crew transitions to E-1, *Loss of Reactor or Secondary Coolant*.
6. RWST Level is approaching the Lo-Lo level setpoint.

Which procedure will address the loss of the “A” SIH Pump; and based on the loss of the “A” SIH Pump, what action is required to be taken?

a) Per E-1, *Loss of Reactor or Secondary Coolant*, the crew will verify adequate sump recirc flowrate with one Charging Pump running.
b) Per E-1, *Loss of Reactor or Secondary Coolant*, the crew will start the swing Charging Pump.
c) Per ES-1.3, *Transfer to Cold Leg Recirculation*, the crew will reopen one RHR Cold Leg Injection Valve.
d) Per ES-1.3, *Transfer to Cold Leg Recirculation*, the crew will restart one RHR Pump.

Proposed Answer: C

Explanation:
This question is considered SRO level, since it requires the applicant to assess plant conditions, determine which procedure addresses the specific failure, and then determine the action to be taken based on those conditions. This goes beyond system knowledge or knowledge of overall procedure strategy.
The crew is currently progressing through E-1, waiting for the RWST Lo-Lo level signal, which will require transition to ES-1.3 to align for cold leg recirculation. Minimum flow assumed by accident analysis is one Charging and one SIH Pump.
“A” and “B” are wrong, since E-1 does not address the loss of SIH Pumps. “A” and “B” are plausible, since E-1 will direct the crew to verify cold leg recirculation capability, and this verifies numerous valves are available to make the alignment. "A" is plausible, since the action of verifying adequate sump recirculation flow with less than one Charging Pump and one SIH Pump is taken in ECA-1.1, *Loss of Sump Recirculation*. "B" is plausible, since two Charging Pumps would provide flow similar to one Charging and one SIH Pump.
"C" is correct, since the RSS pumps utilize a portion of the RHR piping to supply the suctions of the Charging and SIH pumps during the recirculation phase, and ES-1.3 directs opening one RHR cold leg injection valve to provide the flowpath from RSS into the RCS.
“D” is wrong, since ES-1.3 will not direct restarting an RHR Pump. "D" is plausible, since restarting an RHR Pump would increase cooling flow to the core.

Technical Reference(s):	E-1 (Rev. 27), Steps 11-14, and Attachment A
(Attach if not previously provided,	ES-1.3 (Rev. 20), Steps 1-3, especially step 3.p
including version/revision number.)	ESK-5DJ (Rev. 13)

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO, STA) Given a set of plant conditions, determine the required actions to be taken per ES-1.3 and ES-1.4

Question Source:	Modified Bank #407965	
Question History:	Last NRC Exam	N/A
Question Cognitive Level:	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41.5 and 43.5	

Comments:
This question is considered "Modified", since the stem has been changed to have only one train of electrical power available. Another stem condition that has been added is control power has now been lost to the "A" SIH Pump. Distractor "A" has been changed from aligning a hot leg injection path to verifying adequate sump flow through the running Charging Pump.

Original Bank Question #407965

A large break LOCA occurs, and current conditions are as follows:

- No SIH pumps could be started.
- RWST Level is approaching the Lo-Lo level setpoint.

Which procedure will address the loss of SIH Pumps, and based on the loss of SIH Pumps, what action will be taken?

a) Per E-1, *Loss of Reactor or Secondary Coolant*, the crew will verify two Charging Pumps are running.
b) Per E-1, *Loss of Reactor or Secondary Coolant*, the crew will start the swing Charging Pump.
c) Per ES-1.3, *Transfer to Cold Leg Recirculation*, the crew will reopen one RHR Cold Leg Injection Valve.
d) Per ES-1.3, *Transfer to Cold Leg Recirculation*, the crew will restart one RHR Pump.

Answer: C

Examination Outline Cross-reference:	Level	RO	SRO
Question # 89	Tier #		2
K/A Statement: AC Electrical Distribution: Ability to predict the impacts of and use procedures to correct, control, or mitigate the consequences of a Loss of vital AC electrical instrument buses on the AC Electrical Distribution System	Group #		1
	K/A #	062.A2.21	
	Importance Rating		4.3

Proposed Question:

Initial Conditions:

The plant is at 1% power, with preparations being made for entry into MODE 1, when the following sequence of events occurs:

Time:	Event:
T=0:	An Inverter 3 Trouble Annunciator is received on MB8.
T+1 minute:	The BOP operator reports VIAC 3 is deenergized.
T+2 minutes:	The crew dispatches Electrical Maintenance to investigate the problem with Inverter 3.
T+3 minutes:	The crew enters the ACTION for LCO 3.8.3.1.e “Onsite power distribution,” ACTION b, with one AC vital bus not energized.
T+30 minutes:	A PEO successfully energizes VIAC 3 from its alternate source.
T+31 minutes:	The STA looks at the surveillance history for the remaining vital AC buses, and reports Weekly electrical surveillance 4.8.3.1 for VIAC 4 was missed and was last performed 8½ days ago.
T+35 minutes:	The crew is making preparations to complete the missed VIAC 4 electrical lineup surveillance.

In accordance with LCO 3.8.3.1.e (VIAC 3), LCO 3.8.3.1.f (VIAC 4), and Section 3/4.0 of Technical Specifications, what ACTION is the crew required to take?

a) Enter LCO 3.0.3, since LCO 3.8.3.1.e remains in effect for VIAC 3, and LCO 3.8.3.1.f now applies for VIAC 4, since per Surv. Req. 4.0.1, failure to perform a surveillance within the specified interval shall be failure to meet the LCO.
b) Exit LCO 3.8.3.1.e for VIAC 3, and enter LCO 3.8.3.1.f for VIAC 4, since per Surv. Req. 4.0.1, failure to perform a surveillance within the specified interval shall be failure to meet the LCO.
c) Remain in LCO 3.8.3.1.e for VIAC 3, but do not enter LCO 3.8.3.1.f for VIAC 4, since per Surv. Req. 4.0.2, the surveillance time for VIAC 4 has not exceeded its maximum allowable extension.
d) Exit LCO 3.8.3.1.e for VIAC 3, and do not enter LCO 3.8.3.1.f for VIAC 4, but per Surv. Req. 4.0.4, do not enter MODE 1 until the surveillance for VIAC 4 has been completed.

Proposed Answer: C

Explanation: This question is considered SRO level, since it requires the applicant to assess plant conditions, and apply the generic section of Tech Specs (Sections 3.0 and 4.0). Per LCO 3.8.3.1, ACTION b (2), there is a 2-hour requirement to energize a deenergized VIAC.
“B” and “D” are wrong, since when the VIAC was energized from the alternate source, there still is a 24-hour requirement to energize VIAC 3 from its inverter to meet ACTION 3.8.3.1.b (2). “B” and “D” are plausible, since the 2-hour ACTION time to energize VIAC 3 was met. “B” is plausible, since 4.0.1 states failure to meet the surveillance interval shall be failure to meet the LCO, except as provided by 4.0.3. “D” is plausible, since 4.0.4 does prohibit entry into an operational mode unless surveillance requirements are met.
"A" is wrong, and “C” correct, since per Surveillance Requirement 4.0.2, a maximum allowable extension of 1.25 times the surveillance interval of 7 days is 7 days x 1.25 = 8.75 days, and this has not been exceeded (It has been 8.5 days since the surveillance was performed). "A" is plausible, since the surveillance was missed on its scheduled day, and LCO 3.0.3 would be required with both trains INOPERABLE, and there are issues with two opposite train VIACs.

Technical Reference(s): Tech Spec Generic Section, page 3/4 0-1 (Amendment 213)
(Attach if not previously provided, Tech Spec Generic Section, pages 3/4 0-2 and 3 (Amendment 241)
including version/revision number.) Tech Spec LCO 3.8.3.1, page 3/4 8-16 (Amendment 220)
Tech Spec LCO 3.8.3.1 ACTION b, page 3/4 8-17 (Amendment 258)
Tech Spec Surveillance 4.8.3.1, page 3/4 8-17 (Amendment 258)
SP 3670.1-002 (Rev. 21), pages 2 and 3 of 29 (VIAC 3 and 4 Surv.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major administrative or procedural precautions and limitations associated with the 120 VAC Distribution System, including the basis for each, identified within the following… MP3 Technical Specifications …

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 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 90	Tier #		2
K/A Statement: Containment:	Group #		1
Knowledge of surveillance procedures	K/A #	103.G2.2.12	
Proposed Question:	Importance Rating		4.1

The following four pieces of plant equipment have been checked by Surveillance over the past month (Items 1 and 2 were checked during an outage, and items 3 and 4 were checked with the plant online):

1. Aux Building Filter Exhaust Fan outlet dampers 3HVR-AOD196A/B and 197A/B have been verified CLOSED.
2. The ESF Building Porous Concrete Groundwater Sump Pump (3SRW-P5) has been verified FUNCTIONAL.
3. The Reactor Coolant Pump Motor Power Feeder Breaker overcurrent protective devices have been verified FUNCTIONAL.
4. The RHR Suction Relief Valves were determined to be OPERABLE by checking the required RHR suction valves OPEN.

Which three of these surveillance checks were specifically performed to protect the Containment (Primary Containment or Secondary Containment) boundary?

a) 1, 2, and 3
b) 1, 2, and 4
c) 1, 3, and 4
d) 2, 3, and 4

Proposed Answer: A

Explanation:
This question is considered SRO level since it requires the applicant to apply Tech Spec/TRM Basis and Surveillance procedure knowledge to determine the correct answer.
"A" is correct, since the basis for Surveillances 1, 2, and 3 are as follows:

- The basis for verifying the Aux Building Filter Exhaust Fan inlet dampers (3HVR-AOD196A/B and 197A/B closed is to prevent a breach of the SLCRS Boundary.
- The basis for verifying the Containment Recirc Pump Casing is full of RWST water is to prevent Containment sump water from entering the pump. The pump casing is kept at a level of approximately 15.5 feet following dewatering to prevent containment sump water from entering the pump.
- The basis for the verifying the ESF Building Porous Groundwater Sump Pump is Functional is to protect the Containment Steel liner from unanalyzed hydrostatic loading. "C" is plausible, since the ESF Porous Groundwater Sump is part of the ESF Building, which is not directly tied to the Containment structure.
- The basis for verifying the Conductor Overcurrent Protective Devices are Functional is to protect the Containment electrical penetrations and penetration conductors.

"B", "C", and "D" are wrong, since the basis of the RHR Suction Relief Valve Surveillance is to verify the RHR Suction Relief Valves are aligned to provide COPPS, which is required to ensure RCS pressure boundary integrity is not compromised.
"B", "C", and "D" are plausible, since each contains two surveillances whose bases are to protect the Containment Barrier. Also, the RHR Suction Relief Valve surveillance requires the crew to check RHR Suction Valve positions, which are Containment Isolation Valves. Also, none of the three Containment-related surveillances directly check anything tied to the Containment structure or Containment Isolation Valve positions.

Technical Reference(s): SP 3614I.3A (Rev. 1), step 4.1.3.n.
(Attach if not previously provided, OP 3314A (Rev. 27), Precaution 3.10
including version/revision number.) Tech Spec Bases for 3/4.6.6.1 (Revised by NRC Letter A15710)

Tech Spec Bases for 3/4.7.9 (LBDCR 12-MP3-003), page B 3/4 7-23a, second paragraph
TRM Surveillance Requirement 4.6.1.6.3.a (LBDCR 07-MP3-018)
TRM Technical Requirement 3/4.6.1.6.3 Basis (LBDCR 07-MP3-018)
TRM Surveillance Requirement 4.8.4.1 (LBDCR 07-MP3-018)
TRM Technical Requirement 3/4.8.4.1 Basis (LBDCR 07-MP3-018), page 3/4.8-6
TRM Table 3.8.4.1-1 (LBDCR 07-MP3-018), page 3/4.8-13
Tech Spec Surveillance Requirement 4.4.9.3.2 (Amendment No. 258)
Tech Spec Bases for Overpressure Protection Systems, page B 3/4 4-15 (Amend. 204)
Tech Spec Bases for Overpressure Protection Systems, page B 3/4 4-25 (LBDCR 12-MP3-010)

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 containment structure and components and the Containment Leakage Monitoring System components, 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: 55.43.2 and 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 91	Tier #		2
K/A Statement: Nonnuclear Instrumentation: Ability to predict the impacts of and use procedures to correct, control, or mitigate the consequences of Loss of power supply	Group #		2
	K/A #	016.A2.02	
	Importance Rating		3.3

Proposed Question:

Initial Conditions:

- A plant startup is in progress per OP 3203, *Plant Startup*.
- Reactor Power is 7%.

The following sequence of events occurs:

1. Power is lost to Turbine Impulse Pressure instrument 3MSS*PT505.
2. The crew enters AOP 3571, *Instrument Failure Response*.
3. Per AOP 3571, the BOP Operator verifies which instrument has lost power.
4. Per AOP 3571, the RO checks the status of Permissives P-10 and P-13.

What is the most limiting LCO ACTION, if any, currently in effect?

a) No ACTION is required, but 3MSS*PT505 is required to be restored prior to exceeding 10% power.
b) No ACTION is required, but 3MSS*PT505 is required to be restored prior to placing the Main Turbine on-line.
c) Enter LCO 3.3.1, ACTION 8 and place the Bistables associated with 3MSS*PT505 in the tripped condition within 6 hours.
d) Enter LCO 3.0.3 ACTION, and restore 3MSS*PT505 within 1 hour or be in HOT STANDBY within the next 6 hours.

Proposed Answer: D

Explanation:
This question is considered SRO level, since it requires the applicant to assess plant conditions and apply generic LCO requirement 3.0.3.
Loss of power to 3MSS*PT505 causes P-7 and P-13 to be in the wrong state for power less than 10%. Per AOP 3571, the US is required to check permissive status and check Tech Spec requirements. Per Tech Specs, the crew has one hour to check the status of these permissives, but these permissives have already been checked as stated in Item 4 in the question stem.
"D" is correct, and "A", "B", and "C" wrong, since in MODE 1, with these permissives NOT in the required state, the crew is required by LCO 3.3.1, ACTION 8 to enter TS 3.0.3, which requires the condition to be restored within one hour or be in HOT STANDBY within the next six hours.
"A" is plausible, since the P-10 and P-13 Permissives normally change state between 10% and 11% power.
"B" is plausible, since PT505 is used as an indicator of Main Turbine Power.
"C" is plausible, since the normal action requirement in LCO 3.3.1 is to check bistables in the correct condition, depending on how the bistables respond to a loss of power to an instrument, or based on the coincidence associated with these permissives, the permissives could be in the required state.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	AOP 3571 (Rev. 18), Attachment G, steps G.14.a and b
	Tech Spec LCO 3.3.1, Table 3.3-1 (Amend. 266), Functional Unit 17.b
	Tech Spec LCO 3.3.1, Table 3.3-1 (Amend. 266), ACTION 8
	Tech Spec Generic LCO 3.0.3 (Amendment 213)
	Tech Spec Table 2.2-1 (Amendment 242), Functional Unit 18
	Functional Sheet 4 (Rev. G) and Sheet 16 (Rev. M)
	Process Block Sheet 18 (Rev. J)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a plant condition or equipment malfunction (with the Main Steam System), use provided reference material to… Evaluate technical specification applicability and determine required action requirements…

Question Source: Modified Bank #403975
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:
This question is considered "Modified", since in the stem, power has been lost to the instrument, rather than the instrument failing high. This requires the applicant to determine which way the bistables feeding P-10 and P-13 bistables fail on a loss of power to the instrument. Also, distractor "C" has been changed from the Reactor automatically tripping to a requirement to trip bistables in six hours, and distractor "D" has been changed from no ACTION required to a requirement to restore PT505 to service before placing the Main Turbine on-line.

Original Bank Question #403975

Initial Conditions:

- The crew is performing a plant startup.
- Reactor Power is 7%.
- The "A" TDFW pump is running.

The following sequence of events occurs:

1. Main turbine impulse pressure instrument 3MSS*PT505 fails high.
2. The US checks Tech Specs, and directs the RO to verify whether Permissive status is proper for current plant conditions.

Based on the failed instrument and current permissive status, what ACTION, if any, is the crew required to take?

a) Startup may continue, however, PT-505 must be restored prior to the crew increasing power above 10%.
b) Verify the Reactor is tripped and enter Technical Specification 3.0.3.
c) Enter E-0, *Reactor Trip or Safety Injection*.
d) No ACTION required, the startup may proceed.

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Question # 92	Tier #		2
K/A Statement: Waste Gas Disposal: Ability to predict the impacts of and use procedures to correct, control, or mitigate the consequences of RMS alarms and/or malfunctions	Group #		2
	K/A #	071.A2.05	
	Importance Rating		3.1

Proposed Question:

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

1. A RADIATION ALERT and RADIATION HI annunciators come in on MB2.
2. The RO reports the Radioactive Gaseous Waste Radiation Monitor 3GWS-RE48 is in HI ALARM.

Complete the following statement about the automatic action that occurred when the annunciator was received; and which specific procedure actually contains the direction to place the Degasifier in Standby.

The__(1)____ closed; and the US will enter__(2)____ _which will direct the crew to place the Degasifier in Standby.

a) (1) Process Gas Receiver Discharge Pressure Control Valve 3GWS-PV49
   (2) OP 3353.MB2B, 3-9, *Rad Hi*
b) (1) Process Gas Receiver Discharge Pressure Control Valve 3GWS-PV49
   (2) AOP 3573, *Radiation Monitor Alarm Response*
c) (1) Millstone Stack Isolation Dampers 3GWS*AOD78A and B
   (2) OP 3353.MB2B, 3-9, *Rad Hi*
d) (1) Millstone Stack Isolation Dampers 3GWS*AOD78A and B
   (2) AOP 3573, *Radiation Monitor Alarm Response*

Proposed Answer: B

Explanation:
This question is considered SRO level since it requires the applicant to determine which of two applicable procedures—the entry conditions of which both are met, and both of which contain actions related to the abnormal condition in progress—will be most helpful for the event.
"A" and "C" are wrong, since after making some checks, OP3353.MB2B, 3-9 will direct the crew to AOP 3573, which will direct the crew to place the Degasifier in Standby. "A" and "C" are plausible, since the entry condition for OP 3353.MB2B, 3-9 is applicable in this event, and it will direct some actions.
"B" is correct, and "D" is wrong, since the valve that auto-closes on High Radiation as sensed by 3GWS-RE48 is the process gas receiver discharge pressure control valve. "D" is plausible, since these are actual dampers in the Gaseous Waste System that isolate a release path.

Technical Reference(s): OP 3353.MB2B, 3-9 (Rev. 3), Setpoint and steps 1-6
(Attach if not previously provided, AOP 3573 (Rev. 28), Section 2.0, and Attachment A, page 6 of 13
including version/revision number.) P&ID 109B (Rev. 27)
Proposed references to be provided to applicants during examination: None
Learning Objective: Direct crew response to abnormal radiation conditions
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55. 41.7, 43.4, and 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 93	Tier #		2
K/A Statement: Fire Protection:	Group #		2
Conditions and limitations in the facility license	K/A #	086.G2.2.38	
Proposed Question:	Importance Rating		4.5

With the plant initially at 100% power with the Fire Protection Water System in its normal lineup, the following sequence of events occurs:

1. A PEO reports that workers in the vicinity of Charging Pump Water Curtain Sprinkler Header Isolation Valve 3FPW-V553 have inadvertently struck the valve, and it appears to be damaged.
2. Maintenance is sent out to investigate, and they report that 3FPW-V553 has a stem-disk separation, and cannot be opened.
3. The Shift Manager (SM) reviews TRM 3.7.12.2, *Spray and/or Sprinkler Systems* to determine required compensatory actions.
4. The SM is performing a brief with the oncoming compensatory fire watch per C OP 200.17, *Fire Watch and Impairment Tracking*.

Complete the following statement.

A/An__(1)___fire watch patrol is required, whose duties include notifying the Control Room of a fire and warning other personnel in the area, _____(2)_______. **(Reference provided)**

	(1)	(2)
a)	hourly	AND attempting to extinguish the fire if confident of their ability to do so
b)	hourly	BUT NOT attempting to extinguish the fire
c)	continuous	AND attempting to extinguish the fire if confident of their ability to do so
d)	continuous	BUT NOT attempting to extinguish the fire

Proposed Answer: C

Explanation:
This question is considered a KA match, since it deals with compensatory measures for inoperable sprinklers. This question is considered SRO level, since it requires the applicant to interpret TRM requirements, and understand Fire Watch requirements, and briefing the oncoming fire watch is designated as a Shift Manager function.
Per TRM 3.7.12.2.l, the Charging Pump Water Curtain Sprinkler System is required to be FUNCTIONAL. It has a double asterisk (**) after it, which means this water curtain protects areas containing safety-related equipment. ACTION a requires a continuous fire watch to be established for those areas in which both trains of redundant fire safe shutdown systems or components could be damaged, and in other areas, establish an hourly fire watch patrol.
"A" and "B" are wrong, since the Charging Pump Water Curtain Sprinkler System provides separation between the Charging Pumps and the RPCCW Pump area where both trains of safety related equipment can be affected, so a continuous fire watch is required. "A" and "B" are plausible, since if one safety train was affected, an hourly fire watch would be required.
"C" is correct, and “D wrong, since one of the duties for a continuous fire watch is to attempt to extinguish the fire, if the following conditions are met: The Control Room has been notified, other personnel in the area are warned, and the fire watch is confident of their ability to fight the fire.
"D" is plausible, since the fire watch is required to notify the Control Room and warn others in the area first, and should only attempt to fight the fire if they are confident of their ability to extinguish the fire. Millstone has a site fire brigade whose purpose is to fight fires.

Technical Reference(s): TRM 3.7.12.2 (LBDCR 07-MP3-018)
(Attach if not previously provided, C OP 200.17 (Rev. 6), Att. 1, steps 1 and 2, and Att. 2, steps 1 and 2
including version/revision number.) OP 3341A-008 (Rev. 8), page 8 of 31
OP 3341A (Rev 21), Section 4.25
P&ID 146C (Rev. 21)
FPER (Rev. 35), Auxiliary Building Analysis (page 5-20)
FPER (Rev. 35), Response C.6.c.(1) (page B-42) and C.7.k (pages B-59 and 60)
FPW086C (Rev. 6, Ch. 1) Fire Protection Water Lesson Plan, page 12 of 29

Proposed references to be provided to applicants during examination: **TRM 3.7.12.2, pg. 3/4 7-21 and 22**

Learning Objective: Outline the responsibilities of the Shift Manager for Fire Watches

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 43.1 and 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 94	Tier #		3
K/A Statement: Conduct of Operations:	Group #		1
Knowledge of conduct of operations requirements	K/A #	G2.1.1	
Proposed Question:	Importance Rating		4.2

The crew is preparing to station a MRULE (a)(4) dedicated operator, and current conditions are as follows:

- The dedicated operator will be stationed in the area of the Main Boards.
- NO additional levels of 10CFR50.59 evaluations have been completed.
- NO additional NRC approvals have been received.
- The Work Control SRO is performing a pre-job brief with the dedicated operator, and is currently reviewing the dedicated operator's responsibilities.

Complete the following statement in accordance with OP-AA-1500, *Operational Configuration Control*.

When stationed in the area of the Main Boards, the dedicated operator may_____________.

a) be used to perform manual actions to replace Tech Spec listed automatic functions
b) be used to maintain availability of a component and/or system during a testing evolution
c) rely on memory to carry out required actions as long as these actions require less than five steps
d) be the Operator at the Controls if the actions are required to be performed at MB1 through 4

Proposed Answer: B

Explanation:
This question is considered SRO level, since it requires knowledge of administrative procedures that specify to SRO duties.
"A" is wrong, since Tech Spec listed automatic functions shall NOT be replaced with manual actions without prior NRC approval (step 3.2.2). "A" is plausible since this can be allowed if prior NRC approval had been obtained.
"B" is correct, since the dedicated operator may be used to maintain availability of a component and/or system during a testing evolution (step 3.2.1).
"C" is wrong, since the dedicated operator is required to have written procedural guidance (step 3.2.6). "C" is plausible since the procedural guidance is required to contain less than five steps.
"D" is wrong, since if stationed in the control room, the dedicated operator is not allowed to be the Operator at the Controls (step 3.2.4). "D" is plausible, since Main Boards 1 through 4 are the Main Boards monitored by the Operator at the Controls.

Technical Reference(s): OP-AA-1500 (Rev. 21), Section 3.2
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the requirements for using "dedicated operators."

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 # 95	Tier #		3
K/A Statement: Conduct of Operations: Knowledge of the fuel handling responsibilities of SROs (SRO Only)	Group #		1
	K/A #	G2.1.35	
Proposed Question:	Importance Rating		3.9

The plant is in MODE 6, with preparations for core off-load to commence; and current conditions are as follows:

- The Reactor has been subcritical for 120 hours.
- Refueling cavity water level is 25 feet above the Reactor vessel flange.
- The "A" RHR train is INOPERABLE.
- The "B" RHR train is OPERABLE (and in operation).
- RHR flow is 2,700 gpm.

Per OP 3210A, *Refueling Preparation* and 3210B, *Refueling Operations*, which of these conditions would prevent the Refueling SRO from authorizing the commencement of CORE ALTERATIONS?

a) The time the Reactor has been subcritical
b) Refuel Cavity level
c) The number of OPERABLE RHR Trains
d) RHR Pump flow

Proposed Answer: D

Explanation:
This question is considered SRO level since it requires the applicant to know administrative requirements associated with refueling activities.
"A" is wrong, since the Reactor must be subcritical for at least 100 hours.
"B" is wrong, since there must be at least 23 feet of water above the vessel flange.
"C" is wrong, since the requirement is for at least one RHR loop to be OPERABLE and in operation.
"D" is correct, since the required RHR operating range is between 2,800 gpm and 4,000 gpm.
"A", "B", and "C" are plausible, since each of these conditions have a requirement for refueling operations and all of them are close to their limit.

Technical Reference(s):	OP 3210A (Rev. 18), step 4.1.4
(Attach if not previously provided,	OP 3210B (Rev. 14), Prerequisite 2.1.15
including version/revision number.)	OP 3210B (Rev. 14), Precaution 3.9
	Tech Spec LCO 3.9.3 (January 31, 1986)
	Tech Spec LCO 3.9.10 (Amendment 258)

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 #403342
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.43.2, 43.5, and 43.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 96	Tier #		3
K/A Statement: Equipment Control: The process for making	Group #		2
design or operating changes to the facility, such as 10 CFR	K/A #	G2.2.5	
50.59, "Changes, Tests and Experiments," screening and	Importance Rating		3.2

evaluation processes, administrative processes for temporary modifications, disabling annunciators, or installation of temporary equipment

Proposed Question:

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

- Charging line flow control valve 3CHS*FCV121 has failed open.
- The crew has determined they can operate the plant by throttling a manual valve closed in-line with 3CHS*FCV121, and aligning 3CHS*HCV190A to provide a "throttleable" charging flow path.
- This configuration is not controlled by an approved procedure.

How are 10CFR50.59, "*Changes, Tests, and Experiments"* safety evaluation requirements required to be initially addressed?

a) The US will answer questions on OP-AA-100, *Conduct of Operations*, Attachment 9, "Alternate Plant Configuration Sheet".
b) The US will initiate a Condition Report (CR) per PI-AA-200, *Corrective Action*, and the CR process will address the safety evaluation requirements.
c) The SM will evaluate the effects from the alternate plant configuration on WM-AA-301, *Operational Risk Assessment*.
d) The SM will discuss the change with Engineering, and per PI-AA-200, *Corrective Action*, evaluate the effects and log the results in the Operations narrative log.

Proposed Answer: A

Explanation: This question is considered SRO level, since the applicant is required to have knowledge of administrative procedures associated with operating changes in the facility.
"A" is correct, since OP-AA-100, Attachment 6 requires the US to "Determine if engineering assistance and evaluation under 10 CFR 50.59 is needed by completing Attachment 9". Attachment 9, Alternate Plant Configuration Sheet, has a list of 7 questions that must be answered. If yes is answered to any question, then engineering assistance and evaluation per 10 CFR 50.59 is required.
"B" is wrong as the Condition Report (CR) process is not specifically used to determine 10CFR 50.59 applicability. "B" is plausible as a CR is used to help evaluate abnormal or adverse events, and PI-AA-200 provides guidance for Condition Report submittal.
"C" is wrong, since WM-AA-301 does not evaluate or screen for 10 CFR 50.59. "C" is plausible as WM-AA-301 is used to assess and mitigate operational risks associated with plant activities (including emergent activities). "D" is wrong, since the SM does not perform the evaluation and is not required to log results in the Operations narrative log. "D" is plausible, since PI-AA-200 is used to identify deficiencies, and provides direction for Shift Managers to perform Operability Reviews. Also, Engineering will be involved, and the Shift Managers maintain the narrative log.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
OP-AA-100 (Rev. 46), Attachment 6, step 15-2
OP-AA-100 (Rev. 46), Attachment 9
WM-AA-301 (Rev. 24), Table of Contents
PI-AA-200 (Rev. 41) Sections 3.1 and 3.3 and Attachment 3

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the process associated with Alternate Plant Configurations

Question Source: Bank #370369

Question History: Last NRC Exam Millstone 3 2019 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10, 43.3

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 97	Tier #		3
K/A Statement: Equipment Control:	Group #		2
Knowledge of maintenance work order requirements	K/A #	G2.2.19	
Proposed Question:	Importance Rating		3.4

The Work Control SRO is reviewing a Work Order for authorization per WM-AA-100, *Work Management*.

Which of the following would specifically require the Work Control SRO to reject the Work Order?

a) The work requires multiple plant conditions not specifically addressed by the work order.
b) The initiating Condition Report requires multiple plant locations where work will be performed.
c) The work to be carried out is classified as "Minor Maintenance".
d) The initiating Condition Report classifies the work as "Urgent Work".

Proposed Answer: A

Explanation:
This question is considered SRO level, since it requires the applicant to have knowledge of SRO responsibilities while performing administrative procedures during normal plant operations.
"A" is correct, since the SM shall not authorize a Work Order that requires multiple plant conditions (WM-AA-100, step 5.2.23.c, including Caution).
"B", "C", and "D" are wrong, since these conditions do not prevent the SM from authorizing the work order.
"B" is plausible, since if the initiating CR contains multiple locations, a manually generated work order may be required, and one supervisor cannot physically be present in all locations at the same time (WM-AA-100, step 3.5.1, including Note).
"C" is plausible, since work classified as Minor Maintenance allows bypassing of some of the formal planning and scheduling functions (WM-AA-100, step 3.3.3, including Note).
"D" is plausible, since work classified as Urgent Work has specific admirative requirements, and requires an Urgent Work Order (WM-AA-100 (Rev. 37), steps 3.1.10.a-f, including Notes and Caution).

Technical Reference(s): WM-AA-100 (Rev. 37), steps 3.1.10.a-f, including Notes and Caution
(Attach if not previously provided, WM-AA-100 (Rev. 37), step 3.3.3, including Note
including version/revision number.) WM-AA-100 (Rev. 37), step 3.5.1, including Note
WM-AA-100 (Rev. 37), step 5.2.23.c, including Caution

Proposed references to be provided to applicants during examination: None

Learning Objective: (SRO) Describe the conditions required for authorizing release of an AWO.

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55. 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 98	Tier #		3
K/A Statement: Radiation Control: Radiation or contamination	Group #		3
hazards that may arise during normal, abnormal, or emergency	K/A #	G2.3.14	
conditions or activities, such as analysis and interpretation of	Importance Rating		3.8

radiation and activity readings as they pertain to administrative, normal, abnormal, and emergency procedures or to analysis and interpretation of coolant activity, including comparison to emergency plan or regulatory limits (SRO Only).

Proposed Question:

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

1. An unisolable steam break occurs in the Main Steam Valve Building.
2. Automatic and Manual Reactor Trip attempts fail from the Main Boards.
3. The operators successfully shutdown the Reactor.

Current conditions are as follows.

- An ALERT has been declared.
- The CR-DSEO is preparing the Emergency Notification (ENF) Form.
- The CR-DSEO is determining if the "Radiological Release in progress due to event" box is required to be checked on the ENF Form.

The CR-DSEO gathers the following information.

- Turbine Building Stack Monitor 3HVR-RE10A shows an increasing trend, but has NOT reached the ALERT setpoint.
- ESF Building Ventilation Exhaust Monitor 3HVQ-RE49 shows an increasing trend, but has NOT reached the ALERT setpoint.
- Liquid Waste Effluent Monitor 3LWS-RE70 is in ALERT, but is NOT in ALARM.
- A 0.20 GPM tube leak into "D" SG was in progress at the start of the event.

In accordance with MP-26-EPI-FAP06, *Classification and PARs*, which one of these abnormal radiological conditions specifically requires the CR-DSEO to select the "Radiological release in progress due to event" Box?

a) 3HVR-RE10A increasing trend
b) 3HVQ-RE49 increasing trend
c) 3LWS-RE70 in ALERT
d) The "D" SG Tube leak

Proposed Answer: D

Explanation:
This question is considered SRO level since it requires the applicant to interpret radiation readings as they pertain to making an appropriate reporting decision for a specific duty of an SRO licensed individual.
The CRDSEO is required to select that a radiological release to the environment "Is occurring" on the Emergency Notification Form (IRF) due to any of the following:

- Gaseous effluent radiation monitor(s) in ALART or ALARM
- Federal limits being exceeded (a basis of certain Rad Monitor alarm setpoints).
- Primary to secondary leak rate >Tech Specs with a steam release to environment.
- A dose rate to the environment measured downwind within the protected area >0.4 mR/hr (provided by on-shift HP Tech/RMT #1).
- A sample analysis indicating a radioactive liquid release which exceeds federal limits (provided by Chemistry).

“A” and “B” are wrong, since these monitors are not in ALERT or ALARM. “A” and “B” are plausible, since these are gaseous effluent monitors specified in FAP06, Attachment 5, and they have an upward trend. Also, for AOP 3576, tube leakage is verified using either a Radiation Monitor in ALERT, Alarm, or an increasing trend.

"C" is wrong, since this is not gaseous effluent monitor. "C" is plausible, since this monitor is monitoring an effluent point, and it is in ALERT.

“D” is correct, since the Tech Spec limit for leakage into any one SG is 150 gpd, and leakage into the “D” SG (0.20 gallons/minute x 60 minutes/hour x 24 hours/day = 288 gallons per day leakage) has exceeded this limit with a steam release in progress.

Technical Reference(s):	MP-26-EPI-FAP06 (Rev. 13), Section 2.1.8
(Attach if not previously provided,	MP-26-EPI-FAP06 (Rev. 13), Att. 5
including version/revision number.)	Tech Spec 3.4.6.2.c (Amendment No. 238)

Proposed references to be provided to applicants during examination: None

Learning Objective: The Shift Manager and Unit Supervisor will perform all administrative actions necessary to protect the public in accordance with emergency plan procedures.

Question Source:	New
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.43.4 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 99	Tier #		3
K/A Statement: Knowledge of emergency and abnormal operating procedures implementation hierarchy and coordination with other support procedures or guidelines, such as operating procedures, abnormal operating procedures, or severe accident management guidelines.	Group #		4
	K/A #	G2.4.16	
	Importance Rating		4.4

Proposed Question:

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

1. The crew enters AOP 3563, *Loss of DC Bus Power.*
2. Shortly after entering AOP 3563, the Reactor trips.

The US is considering how the crew will implement E-0, *Reactor Trip or Safety Injection*, ES-0.1, *Reactor Trip Response*, and AOP 3563 to mitigate the event now that the reactor has tripped.

Which procedure implementation strategy complies with the guidance in OP 3272, *EOP User's Guide* for this situation?

a) Prior to entering E-0, the US directs the RO to complete a step in AOP 3563 that assists with RCS temperature control by reading the step out-loud to the RO. After exiting E-0, the remaining steps in AOP 3563 are performed in parallel with ES-0.1.
b) After exiting E-0, the US directs the RO and the BOP by reading out-loud all of the steps of AOP 3563. After the crew completes all of AOP 3563, the crew then enters ES-0.1.
c) After exiting E-0, the US directs the RO and the BOP by reading out-loud steps from ES-0.1 and AOP 3563 in parallel. Only the steps of AOP 3563 that are necessary to ensure success of ES-0.1 are performed, without completing all of AOP 3563.
d) Prior to entering E-0, the US hands AOP 3563 to the extra SRO, who directs the BOP by reading out loud the steps of AOP 3563, while the US directs the RO to perform the immediate actions of E-0, and then transitions to and performs ES-0.1.

Proposed Answer: C

Explanation:
This question is considered acceptable for Tier 3, since the specific EOP and AOP listed in the question are not being tested. They are included to enhance the operational validity of the question. This question is considered SRO level, since applicants are required to have knowledge of administrative procedures that specify coordination of plant emergency and abnormal procedures.
"A" is wrong, since the actions of an AOP are not to be initiated before completing all immediate actions of the ERG (Westinghouse Emergency Response Guideline) procedure, and E-0, steps 1 through 4 are immediate action steps. "A" is plausible, since actions of an AOP are allowed to be performed in parallel with EOPs.
"B" is wrong, since the actions of the ERG procedure are required to receive priority over the AOP, so it is not acceptable to complete the AOP before entering the required ERG procedure, since the AOP actions are to be implemented on a "not to interfere basis with the EOP. "B" is plausible, since actions of an AOP are allowed to be performed in parallel with EOPs.
"C" is correct, since it is acceptable to perform the actions of an AOP in parallel with an ERG derived EOP provided the actions of the ERG-derived procedure receives priority and the actions of the AOP are not initiated before completing all immediate actions of the ERG derived procedure. It is not necessary to perform all steps in the parallel procedure; only those steps necessary to ensure success of the ERG derived procedure need to be performed.
"D" is wrong, since the actions of an AOP are not to be initiated before completing all immediate actions of the ERG procedure, and E-0, steps 1 through 4 are immediate action steps. "D" is plausible, since actions of an AOP are allowed to be performed in parallel with EOPs.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP3272 (Rev. 14), Section 3.1, page 4, bottom paragraph OP3272 (Rev. 14), Section 3.1, page 5, top paragraph OP3272 (Rev. 14), Section 3.8, esp. page 21, top two paragraphs E-0 (Rev. 36), steps 1-4

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:	Bank #408104
Question History:	Last NRC Exam Millstone 3 2015 NRC Exam
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 # 100	Tier #		3
K/A Statement: Emergency Procedures / Plan:	Group #		4
Knowledge of SRO responsibilities in emergency plan	K/A #	G2.4.40	
implementing procedures (SRO Only)	Importance Rating		4.5

Proposed Question:

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

1. An ALERT is declared at Millstone 3.
2. The SM becomes the CR-DSEO.
3. One hour later, the CR-DSEO completes the turnover with the EOF-DSEO.
4. The CR-DSEO becomes the MCRO.

Now that the turnover with the EOF DSEO is complete, which of the following is now a responsibility of the MCRO per EPI-FAP01, *Control Room Emergency Operations*?

a) Provide current plant status to the ADTS
b) Classifying Events
c) Take full control of RMT #1 members
d) Make offsite notifications

Proposed Answer: A

Explanation:
This question is considered SRO level, since it requires the applicant to have knowledge of SRO responsibilities during EPlan activation.
Per FAP01, step 5.2.2, MCRO responsibilities include the following:

- Recommend corrective actions to the ADTS
- Provide current plant status to the ADTS
- Recommend classification changes to the ADTS
- Coordinate actions to mitigate degradation of plant systems with the ADTS.
- Coordinate Control Room actions and repair team activities with the MOSC

"A" is correct, since the MCRO reports directly to the ADTS (FAP01, step 2.2), and is responsible for providing current plant status to the ADTS (FAP01, step 5.2.2.b).
"B" is wrong, since this is not a responsibility of the MCRO. "B" is plausible, since this was a responsibility of the SM while performing the role of the CR-DSEO (FAP01, step 5.2.1.b), and the MCRO is now responsible for recommending event classification changes to the ADTS (FAP01, step 5.2.2.c).
"C" is wrong, since this is not a responsibility of the MCRO. "C" is plausible, since this is a responsibility of the MRCA (FAP01, step 2.7).
"D" is wrong, since this is not a responsibility of the MCRO. "D" is plausible, since this is a responsibility of the Emergency Communicator (FAP01, step 5.2.4).

Technical Reference(s): EPI-FAP01 (Rev. 8), steps 2.2, 2.7, 5.2.1, 5.2.2.b, 5.2.2.c, and 5.2.4
(Attach if not previously provided,
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Explain the role of the Manager of Control Room Operations (MCRO) following the turnover with the EOF DSEO

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.43.5
Comments: