REACTOR OPERATOR WRITTEN EXAM KEY- 2013 EXAM
MILLSTONE UNIT 3

1. D
2. B
3. A
4. D
5. C
6. D
7. B
8. A
9. D
10. C
11. B
12. D
13. D
14. D
15. D
16. A
17. B
18. C
19. B
20. A
21. B
22. A
23. C
24. D
25. D
26. A
27. A
28. A
29. B
30. D
31. A
32. C
33. A
34. C
35. D
36. D
37. C
38. B
39. B
40. C
41. B
42. C
43. A
44. B
45. A
46. C
47. D
48. C
49. D
50. A
51. D
52. C
53. C
54. D
55. C
56. A
57. C
58. B
59. B
60. A
61. D
62. C
63. C
64. A
65. B
66. A
67. A
68. C
69. D
70. B
71. B
72. D
73. B
74. B
75. C

SENIOR REACTOR OPERATOR WRITTEN EXAM KEY- 2013 EXAM
MILLSTONE UNIT 3

1.	D	33.	A	65.	B	97.	D
2.	B	34.	C	66.	A	98.	B
3.	A	35.	D	67.	A	99.	A
4.	D	36.	D	68.	C	100.	D
5.	C	37.	C	69.	D		
6.	D	38.	B	70.	B		
7.	B	39.	B	71.	B		
8.	A	40.	C	72.	D		
9.	D	41.	B	73.	B		
10.	C	42.	C	74.	B		
11.	B	43.	A	75.	C		
12.	D	44.	B	76.	B		
13.	D	45.	A	77.	B		
14.	D	46.	C	78.	B		
15.	D	47.	D	79.	D		
16.	A	48.	C	80.	A		
17.	B	49.	D	81.	D		
18.	C	50.	A	82.	C		
19.	B	51.	D	83.	A		
20.	A	52.	C	84.	A		
21.	B	53.	C	85.	C		
22.	A	54.	D	86.	C		
23.	C	55.	C	87.	B		
24.	D	56.	A	88.	D		
25.	D	57.	C	89.	A		
26.	A	58.	B	90.	A		
27.	A	59.	B	91.	B		
28.	A	60.	A	92.	A		
29.	B	61.	D	93.	B		
30.	D	62.	C	94.	C		
31.	A	63.	C	95.	D		
32.	C	64.	A	96.	C		

Examination Outline Cross-reference:	Level	RO	SRO
Question # 1	Tier #	1	1
K/A Statement: Reactor Trip Stabilization:	Group #	1	1
Interrelation with Reactor Trip Status Panel	K/A #	EPE.007.K2.03	
Proposed Question:	Importance Rating	3.5	3.6

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

1. Power Range NIS Channel N43 (Protection Set 3) drifts down to 50%.
2. The crew enters AOP 3571, *Instrument Failure Response*.
3. The US directs the RO to "Check the existing bistable status to ensure a reactor trip will not occur when the failed channel is tripped."

Which Channel 1 (Red) bistable status light on Main Board 4, if illuminated, would indicate that an automatic reactor trip would occur if the bistables associated with NIS Channel N-43 were tripped?

a) Source Range Hi Counts.

b) Intermediate Range Hi Current.

c) OPΔT.

d) OTΔT.

Proposed Answer: D

Explanation (Optional): "D" is correct since the Channel 1 OTDT bistable needs to be tripped, since PRNIs input to OTDT, with a 2/4 coincidence. "C" is wrong, since PRNIs do not input to OPDT, but plausible, since NIS does input to OTDT. "A" and "B" are wrong, since P-10 will not change state (2/4 coincidence) since 3 channels going below 10% are required to clear P-10. "A" and "B" are plausible, since P-10 is affected by PRNIs, and both SR and IR trips have a 1/2 coincidence and receive input from P-10.

Technical Reference(s): AOP 3571 (Rev 9-7), Att. D, step 4.c and 5 (Attach if not previously provided)
AOP 3571(Rev 9-7), Att. D, Table, page 7 of 7
Functional Dwgs 3 (Rev. G), 4 (Rev. G), and 5 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05493 Describe the operation of the following RPS controls and interlocks… Reactor Trip Signals (As available)

Question Source: Bank 85287

Question History:

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 # 2	Tier #	1	1
K/A Statement: Small Break LOCA:	Group #	1	1
Reasons for actions contained in the EOP	K/A #	EPE.009.EK3.21	
Proposed Question:	Importance Rating	4.2	4.5

The plant has tripped due to a small break LOCA. The crew is currently in ES-1.2, *Post LOCA Cooldown and Depressurization*, and the following conditions exist:

- Pressurizer Level is 18% and steady.
- RCS hot leg temperature is 365°F and steady.
- The crew is checking whether they should isolate SI accumulators.
- The crew notes that the RCS does NOT have adequate subcooling to isolate the accumulators.
- ES-1.2 directs the crew to isolate SI accumulators in spite of the lack of subcooling, since RCS hot leg temperature is less than 440°F.

Why is the crew directed to isolate accumulators without adequate subcooling?

a) RCS hot leg temperature has dropped to where a pressurized thermal shock concern exists. The SI accumulators are isolated to allow the RCS to continue to depressurize, and prevent cold accumulator water from entering the RCS Cold Legs.

b) RCS hot leg temperature has dropped to less than the point corresponding to accumulator pressure after water discharge. The contents have been discharged, and isolation prevents accumulator nitrogen injection into the RCS.

c) RCS hot leg temperature is less than the saturation temperature corresponding to RHR pump discharge pressure. The RHR pump running in the injection mode will ensure RCS subcooling is maintained after the SI accumulators are isolated.

d) RCS hot leg temperature is at the saturation temperature corresponding to accumulator injection pressure. The SI accumulators are isolated to allow RCS pressure to drop to the point where RHR pumps will be able to inject.

Proposed Answer: B

Explanation (Optional): Adequate subcooling insures that accumulator injection is not required. Below 440°F with inadequate subcooling, RCS pressure has dropped to the accumulator pressure AFTER accumulator water has injected. If accumulators are not isolated, nitrogen will inject, producing either a "hard" bubble in the Pressurizer or gas binding in the SG U-Tubes ("B" correct, "A", "C", and "D" wrong). "A" is plausible since the RCS has cooled down, and FR-P.1 terminates SIS to prevent RCS repressurization, and starts an RCP to mix warm RCS water with cold injection water. "C" is plausible since this is the basis for stopping a Charging pump with inadequate subcooling if RCS hot legs are below 340°F. "D" is plausible, since accumulator injection holds up RCS pressure, and is related to the basis for isolating accumulators if adequate subcooling exists.

Technical Reference(s): ES-1.2 (Rev 18-0), step 22; WOG Bkgd Doc (Rev 2) for ES-1.2, step 23 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05530 Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 ES-1.2. (As available)

Question Source: Bank 77419

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 3	Tier #	1	1
K/A Statement: RCP Malfunctions:	Group #	1	1
Determine/interpret abnormalities in RCP air vent	K/A #	APE.015/017.AA2.02	
path and/or oil cooling system	Importance Rating	2.8	3.0

Proposed Question:

With the plant operating at 100% power, the "B" Reactor Coolant Pump develops a tube leak in its Upper Oil Reservoir Cooler.

Correctly complete the following statement regarding Reactor Plant Component Cooling Water (RPCCW), Reactor Plant Chilled Water (CDS), and lubrication to the "B" RCP.

Lubrication will be lost due to___(1)____leaking into the___(2)____.

	(1)	(2)
a)	RPCCW	Oil Reservoir
b)	CDS	Oil Reservoir
c)	Lube Oil	RPCCW System
d)	Lube Oil	CDS System

Proposed Answer: A

Explanation (Optional): "A" is correct since RPCCW cools the RCP oil coolers and its pressure is above oil reservoir pressure, so on a cooler tube leak, RPCCW will enter the oil cooler, lowering the oil's lubrication capability. "B" and "D" are wrong since CDS does not supply the upper oil reservoir cooler. "B" and "D" are plausible since both RPCCW and CDS cool Containment loads, and CDS does cool the RCP motor coolers. "C" is wrong since RPCCW pressure is greater than oil cooler pressure. "C" is plausible since this would occur if oil cooler pressure was higher than RPCCW pressure, and this is the case for several RPCCW loads.

Technical Reference(s): OP3353.MB4B (Rev 4-12), 4-2A (Attach if not previously provided)
P&ID 102A (Rev 31)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05431 Describe the operation of the RCPs under the following abnormal conditions... Conditions requiring a Manual RCP Trip... (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3 and 41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 4	Tier #	1	1
K/A Statement: Loss of RHR System:	Group #	1	1
Reasons for isolating RHR low pressure	K/A #	APE.025.AK3.02	
piping prior to raising pressure	Importance Rating	3.3	3.7

Proposed Question:

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

- The "A" Train of RHR is in operation.
- The "B" Train of RHR is aligned for injection.

The following sequence of events occurs:

1. Annunciator MB2C, 1-6, RHR A SUCT VLV OPEN AND RCS PRESS HI illuminates.
2. Per the annunciator response procedure, the crew stops the "A" RHR Pump (3RHS*P1A).
3. Per the annunciator response procedure, the crew closes 3RHS*MV8701B, "A ISOL (OUT)".
4. The crew transitions to EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory*.

Why is the crew required to close 3RHS*MV8701B prior to transitioning to EOP 3505?

a) Ensure COPPS remains available while aligning the "A" Train of RHR for injection.

b) Maintain letdown flow while aligning the "A" Train of RHR for injection.

c) Prevent the RHR Pump Suction Piping from exceeding 260 °F.

d) Protect the RHR System piping from high RCS pressures.

Proposed Answer: D

Explanation (Optional): The RHS is designed to be isolated from the RCS whenever the RCS pressure exceeds the RHS design pressure of 600 psig, and this action is directed per the ARP with RCS pressure at 765 psia (750 psig). The RHS is isolated from the RCS on the suction side by three normally closed, motor-operated valves in series on each suction line. Two of the motor-operated valves are interlocked to prevent its opening if RCS pressure is greater than 412.5 psia and alarm in the control room if RCS pressure exceeds 440 psig and the valve is open. If the plant is in Mode 1, 2, or 3, the operator is required to close all three suction valves. Per the FSAR, if the plant is in mode 4, 5, or 6 and the RCS pressure increases to 750 psig, the operator is required to close the motor-operated valve closest to the pump ("D" correct). "A" is wrong, since COPPS is also available through the Pzr PORVs, and these actions will isolate the RHR relief valves from the RCS. "B" is wrong, since the letdown from RHR will be isolated by these actions, but will be maintained through the CHS letdown orifices. "C" is wrong, since RHR is not required to be available for ECCS in MODE 5. "A" and "B" are plausible, since part of lining up a train of RHR for injection is to isolate its suction from the RCS, and RHR is used for COPPS and provides a letdown path in MODE 5. "C" is plausible, since RCS cooling has been lost, and on a cooldown, the crew delays placing the second RHR Train in the cooldown mode until RCS is less than 260°F to avoid boiling in the suction line if it has to be aligned for injection.

Technical Reference(s): OP 3353.MB2C (Rev 0-0), 1-6, steps 5.1-5.3 (Attach if not previously provided)
FSAR, Table 5.4-8 (Rev 16)
FSAR 5.4.7.1 (Rev 21.3) Page 5.4-23 and 24

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05457 Describe the major administrative or procedural precautions and limitations placed on the operation of the Residual Heat Removal system, including the basis for each. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 5	Tier #	1	1
K/A Statement: Loss of Component Cooling Water:	Group #	1	1
Ability to verify alarm setpoints and operate	K/A #	APE.026.GEN.2.4.50	
controls per the alarm response manual	Importance Rating	4.2	4.0

Proposed Question:

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

1. The RPCCW SURGE TK LEVEL HI annunciator is received on MB1C.
2. The crew enters the appropriate Annunciator Response Procedure.
3. The US directs the RO to compare the two train-related Surge Tank level indications (3CCP-LI20A and B) on Main Board 1 (MB1).

Correctly complete the following statement regarding the crew's ability to diagnose and control the event.

The crew___(1)___be able to determine the affected train at MB1, and a control switch___(2)___ available at MB1 to manually close the Surge Tank Fill Valve (3CCP-LV20).

	(1)	(2)
a)	WILL	IS
b)	WILL	IS NOT
c)	WILL NOT	IS
d)	WILL NOT	IS NOT

Proposed Answer: C

Explanation (Optional): This question tests the second part of the KA Statement. There is a divider plate separating the two trains of RPCCW in the surge tank ("A" and "B" plausible), but the divider plate only goes up to about 88%, which is below the high level alarm setpoint, so both trains will indicate the same ("A" and "B" wrong). "C" is correct, and "D" wrong, since the Surge Tank Fill Valve has both automatic and manual controls and indication available at MB1. "D" is plausible, since numerous indications exist on the Main Boards without corresponding controls.

Technical Reference(s): OP 3353.MB1C (Rev 6-3), 2-7A, steps 1 and 2 (Attach if not previously provided)
P&ID 121A (Rev 32)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04150 Describe the operation of the following RPCCW System equipment controls and interlocks: Surge Tank Makeup Control Valve... (As available)

Question Source: New

Question History:

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 # 6	Tier #	1	1
K/A Statement: Pressurizer Pressure Control Malfunction:	Group #	1	1
Determine/interpret letdown flow indication	K/A #	APE.027.AA2.08	
Proposed Question:	Importance Rating	3.2	3.2

Initial Conditions:

- A Reactor Startup is in progress per OP 3202, *Reactor Startup.*
- Reactor power is stable at $1x10^{-8}$ Amps in the intermediate range.

The RO reports that a Pressurizer Pressure Controller malfunction has resulted in Master Pressure Controller 3RCS*PT455 output failing to 0%.

Assuming no operator actions are taken, LETDOWN flow through Letdown Isolation Valve 3RCS*LCV459 will initially__________.

a) increase and then stabilize at a higher equilibrium value

b) increase and then isolate

c) decrease and then stabilize at a lower equilibrium value (greater than 0 gpm)

d) decrease and then isolate

Proposed Answer: D

Explanation (Optional): Controller output failing low will generate a "decrease pressure" signal, causing Pzr Spray Valves to open, resulting in actual pressure decreasing. This lowers DP across the letdown line, causing an initial decrease in letdown flow ("A" and "B" wrong). "A" and "B" are plausible, since RCS pressure would increase if the master controller output failed to 100%. "D" is correct and "C" wrong, since backup heaters will not energize on lowering pressure with the master pressure controller output failed to 0%, and Pzr Low Pressure SI is armed (1892 psia) since during a reactor startup, RCS pressure is above P-11 (2000 psia). When SI actuates, a CIA signal is generated, and letdown isolates. "C" is plausible, since the letdown flow starts decreasing as RCS pressure drops, but it will not stabilize. Also, the letdown line relief valve would lift if letdown were isolated downstream of the relief valve, but one of the letdown Containment Isolation Valves is upstream of this valve.

Technical Reference(s): Functional Dwgs 6 (Rev H) and 11 (Rev H) (Attach if not previously provided)
P&ID 104A (Rev 54)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05342 Given a failure, partial or complete, of the Pressurizer Pressure and Level Control System, determine the effects on the system and on interrelated systems. (As available)

Question Source: New

Question History:

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 # 7	Tier #	1	1
K/A Statement: ATWS:	Group #	1	1
Determine/interpret Rod Step Counters and RPI	K/A #	EPE.029.EA2.08	
Proposed Question:	Importance Rating	3.4	3.5

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

1. The "D" MSIV spuriously closes.
2. The RO reports the receipt of a first-out reactor trip annunciator on MB4.
3. The reactor does NOT trip when the RO and BOP attempt to trip the reactor using the Reactor Trip switches.
4. The BOP attempts to open the supply breakers to 480V Load Center 32B, and the breakers do NOT open.
5. The BOP successfully opens the supply breakers to 480V Load Center 32N.

Assuming no further operator action has been taken, correctly complete the following statement regarding status of the Rod Group Demand Counters and DRPI indication for all Shutdown Bank Rods on MB4.

The Group Demand Counters indicate fully___(1)____, and DRPI indicates fully___(2)____at MB4.

	(1)	(2)
a)	inserted	inserted
b)	withdrawn	withdrawn
c)	inserted	withdrawn
d)	withdrawn	inserted

Proposed Answer: B

Explanation (Optional): The reactor trip breakers are still closed, and even though Load Center 32N has deenergized ("A", "C", and "D" plausible), 32B is still closed, and this Load Center provides a parallel path to the Control Rods via parallel Rod Drive MG Sets. So the reactor has not tripped, and the crew will transition to FR-S.1, *Response to Nuclear Power Generation/ATWS*. "B" is correct, since DRPI will continue to show the rods fully withdrawn ("A" and "D" wrong), and there has been no demand signal sent to the shutdown rods ("C" wrong).

Technical Reference(s): FR-S.1 (Rev 19-0), Entry Conditions (Attach if not previously provided)
E-0 (Rev 27-0), step 1, including Note
ES-0.1 (Rev 25-0), step 6

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04624 Identify plant conditions requiring entry into EOP 35 FR-S.1. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.6, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 8	Tier #	1	1
K/A Statement: Steam Gen. Tube Rupture:	Group #	1	1
Knowledge of limiting conditions for	K/A #	EPE.038.GEN.2.2.22	
operations and safety limits	Importance Rating	4.0	4.7

Proposed Question:

The plant is operating at 75% power and the latest leak rate data shows:

- 7.20 GPM Total RCS leakage rate
- 4.50 GPM Leakage into the Pressurizer Relief Tank
- 1.80 GPM Leakage into the Containment Drains Transfer Tank
- 0.20 GPM Leakage into "A" SG
- 0.05 GPM Leakage into "B" SG
- 0.05 GPM Leakage into "C" SG
- 0.10 GPM Leakage into "D" SG

Which of the following RCS Leakage Technical Specifications, if any, have been exceeded?

a) Primary to Secondary

b) Unidentified

c) Identified

d) None, all leakage limits are met

Proposed Answer: A

Explanation (Optional): Reactor to secondary leakage is included as Identified Leakage, and its total leakrate into all four SGs is 0.4 gpm. However, the leakage from the "A" SG is 288 gpd (0.20 x 60 x 24) which is greater than the 150 gpd allowed ("A" correct, "D" wrong, but plausible). Identified leakage is at least 6.7 gpm (PRT + CDTT + Primary to Secondary), and total leakage is 7.2, meaning identified leakage is less than 10 gpm ("C" wrong, but plausible). Total leakage minus identified leakage is at most 7.2 gpm – 6.7 gpm = 0.5 gpm, which is less than the 1 gpm unidentified allowed ("B" wrong but plausible).

Technical Reference(s): Tech Spec 3.4.6.2 (Amend 238)
Tech Spec Defn 1.16 .2, Page 1-3 (Amend 246)
Tech Spec Defn 1.16.2.c and 1.16.4(Amend 238)
(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05444 Describe the major administrative or procedural precautions and limitations placed on the operation of the Reactor Coolant System… (As available)

Question Source: Bank 80098

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 9	Tier #	1	1
K/A Statement: Loss of Main Feedwater:	Group #	1	1
Ability to perform specific system and integrated	K/A #	APE.054.GEN.2.1.23	
plant procedures during all modes of operation	Importance Rating	4.3	4.4

Proposed Question:

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

1. The BOP Operator reports main feedwater flow is decreasing.
2. The TDFW PP A/B EAP CONST SIG LOST annunciator comes in on MB5.
3. The crew references the ARP.
4. The US directs the BOP to take manual control of the "A" TDMFP.

How will the BOP operator regain manual control of the "A" TDMFP?

a) The TDMFP Master Speed Controller (3FWS-SK509A) is taken to MANUAL and used to adjust TDMFP "A" speed.

b) The "PP A SPEED CNTL" NUS Controller for TDMFP "A" (3FWS-SK46A) is taken to MANUAL and used to adjust TDMFP "A" speed.

c) The Manual Speed Controller (3TFC-M1A) is lowered until TDMFP speed starts to decrease, and then the "PP A SPEED CNTL" NUS controller for TDMFP "A" (3FWS-SK46A) is taken to MANUAL and raised to the high speed stop.

d) The Manual Speed Controller (3TFC-M1A) is lowered until TDMFP speed starts to decrease, and the hydraulic jack for TDMFP "A" is placed to "ON"

Proposed Answer: D

Explanation (Optional): This question is considered a KA Match since the actions taken by the BOP operator describe how specific procedure steps are required to be performed. This annunciator comes in when the Westinghouse speed control signal is lost, and locks the EAP controllers for both TDMFPs at the "last called for" signal. "A" and "B" are wrong, since both of these require the EAP controller to respond, either to the input from the master controller, or from manual control. "D" is correct, and "C" is wrong, since TDMFP speed is controlled by the lower speed setting of either 3FWS-SK46A "TD FW A" "PP A SPEED CNTL," or the Manual Speed Controller, and 3FWS-SK46A "TD FW A" "PP A SPEED CNTL," cannot be raised out of the way manually with this failure unless the hydraulic jack is used. The hydraulic jack blocks the hydraulic bleed path from the EAP positioner, driving it to the high speed stop. "A", "B", and "C" are plausible, since all of these controllers work together to control feed pump speed.

Technical Reference(s): OP 3353.MB5C (Rev 4-7), 4-6 (Attach if not previously provided)
LSK 6-1.2E (Rev 10)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04660 Describe the operation of the following Main Feedwater & Steam Generator Water Level Control Systems Controls & Interlocks… Turbine Driven Main Feed Pump Manual Speed Controllers (TFC-M1A/B), Turbine Driven Feed Pump Speed Controllers (FWS-SK46A/B), Turbine Driven Main Feed Pump Master Speed Controller (FWS-SK509A), Turbine Driven Main Feed Pump Hydraulic Jack… (As available)

Question Source: Bank 75608
Question History:
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 10	Tier #	1	1
K/A Statement: Station Blackout:	Group #	1	1
Knowledge of setpoints, interlocks and automatic	K/A #	EPE.055.GEN.2.4.2	
actions associated with EOP entry conditions	Importance Rating	4.5	4.6

Proposed Question:

A total loss of AC power occurs and the following sequence of events occurs:

1. The crew enters ECA-0.0, *Loss of All AC Power.*
2. After about 25 minutes, the "A" Emergency Diesel is locally started.
3. Power is restored to the "A" Train Emergency Bus.

Current conditions are as follows:

- Highest Core Exit TC: 552°F.
- RCS Pressure: 1340 psia.
- Pressurizer Level: 18%.

What procedural action is required to be taken by the crew?

a) Remain in ECA-0.0, *Loss of All AC Power* and commence a rapid cooldown of the RCS.

b) Transition to ECA-0.1, *Loss of All AC Power - Recovery Without SI Required.*

c) Transition to ECA-0.2, *Loss of All AC Power - Recovery With SI Required.*

d) Transition to ES-0.2, *Natural Circulation Cooldown.*

Proposed Answer: C

Explanation (Optional): Considered acceptable as an RO level question since Millstone 3 has an associated objective, and the conditions align directly with the transition step in the procedure, testing overall strategy of ECA-0.2; dealing with loss of subcooling or RCS inventory. "C" is correct, and "A", "B", and "D" wrong, since transition to ECA-0.2 is required for either of the following:

- CETC less than 32 °F subcooling (Subcooling is about 28 °F)
- Pressurizer level is less than 16% (Pzr level is 18%)

If none of the requirements are met, then transition to ECA-0.1 is appropriate ("B" plausible). "A" is plausible, since only one bus has been restored, and ECA-0.0 conducts a rapid cooldown of the RCS if no Emergency Busses are restored. "D" is plausible, since ES-0.2 would be transitioned to after completion of ECA-0.1 or ECA-0.3, to cooldown the RCP seals and take the plant to cold shutdown.

Technical Reference(s): ECA-0.0 (Rev 23-0), Step 30 (Attach if not previously provided)
Steam Tables

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

Learning Objective: MC-03860 Identify plant conditions requiring entry into EOP 35 ECA-0.2 (As available)

Question Source: Modified Bank 64322 (Parent question attached)

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Bank Question 64322 (prior to modification) is attached on the following page.

Examination Outline Cross-reference:	Level	RO	SRO
Question # 11	Tier #	1	1
K/A Statement: Loss of Off-site Power:	Group #	1	1
Determine/interpret Sequencer status lights	K/A #	APE.056.EA2.38	
Proposed Question:	Importance Rating	3.7	3.8

Initial Conditions:

- The plant is at 100% power.
- The RO is at the "A" Train Sequencer Cabinet taking daily rounds.

The following sequence of events occurs:

1. Offsite power is lost.
2. Thirty-nine (39) seconds after the EDGs energize the Emergency Busses, the RO observes the following red lights lit at the "A" Sequencer:
   - The "A" RHR Pump LOP Trip Signal Light.
   - The "A" MDAFW Pump LOP Trip Signal Light.
   - The "A" (Lead) SWP Pump Sequenced Safeguard Start (SSS) Signal Light.
   - The "A" RPCCW Pump Sequenced Safeguard Start (SSS) Signal Light.

Which of these lit red status lights is UNEXPECTED?

a) The "A" RHR Pump LOP Trip Signal Light

b) The "A" MDAFW Pump LOP Trip Signal Light

c) The "A" (Lead) SWP Pump SSS Signal Light

d) The "A" RPCCW Pump SSS Signal Light

Proposed Answer: B

Explanation (Optional): The sequencer will initiate an LOP-Only bus strip and load sequence. All 4KV Pumps except for the running Charging Pump will initially strip. The lead SWP pump will start at about 20 seconds, the RPCCW Pump will start at about 25 seconds, and the MDAFW Pump will start at about 30 seconds. "A" is wrong, since the RHR Pump received a strip signal, and not a start signal, and the LOP Trip Red light means the trip relay has operated as expected. "C" and "D" are wrong, since these pumps should have already started, and the SSS red light means the start signal output relay has actuated as expected. "B" is correct, since the MDAFW Pump should have received an auto-start signal at 30 seconds, and the Red light means the LOP Trip logic is still met, which is unexpected. "A", "C", and "D" are plausible, since each of these lights are lit at some point in the LOP/SIS strip/load sequence.

Technical Reference(s): LSK 24-9.4A (Rev 12) (Attach if not previously provided)

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Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04412 Describe the operation of the following Emergency Diesel Load Sequencer controls and interlocks… SSS/MTB and LOP Trip split indicators… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 12	Tier #	1	1
K/A Statement: Loss of Vital AC Elec. Inst. Bus	Group #	1	1
Reasons for actions contained in the EOP	K/A #	APE.057.K3.01	
Proposed Question:	Importance Rating	4.1	4.4

With the plant at 100% power, VIAC 1 is lost and the crew enters AOP 3564, *Loss Of One Protective System Channel.*

AOP 3564, step 3 directs the crew to defeat the failed channel input and align the "Loop Temp Cutout – Tavg" Switch (MB4) to "Loop A" and pull out.

One reason AOP 3564 directs the crew to take this action is to block Loop A Tave input to ________.

a) P-12 to Steam Dumps

b) C-3 to Turbine Runback

c) C-4 to Turbine Runback

d) C-16 to Block Turbine Load

Proposed Answer: D

Explanation (Optional): VIAC 1 feeds vital instruments. Loss of VIAC 1 fails Loop A input to the Tave Hi and Lo Auctioneering Circuit. The Auctioneered Hi Circuit inputs to Steam Dumps, Pressurizer Level Control, Rod Insertion Limit alarms (Rod Lo and Lo-Lo alarms). The Auctioneered Lo Circuit inputs to C-16 Block Turbine Loading circuit ("D" correct). "A", "B", and "C" are wrong, since each of these permissives/controls receive a 2 of 4 coincidence logic that does not receive an input from the Loop Temp Cutout Switch. "A", "B", and "C" are plausible, since each of these receive input from Loop A RCS Tave.

Technical Reference(s): AOP 3564 (Rev 10-0), step 3 (Attach if not previously provided)
Functional Dwg 5 (Rev K)
Functional Dwg 9 (Rev H)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03956 Discuss the basis of major precautions, procedure steps and/or sequence of steps within AOP 3564, Loss of One Protective System Channel. (As available)

Question Source: Bank 70374

Question History:

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 # 13	Tier #	1	1
K/A Statement: Loss of DC Power:	Group #	1	1
Operational implications of Battery Charger	K/A #	APE.058.AK1.01	
equipment and instrumentation	Importance Rating	2.8	3.1

Proposed Question:

With the plant initially at 100% power, DC Bus 4 deenergizes, resulting in the following sequence of events:

1. The crew enters AOP 3563, *Loss of DC Bus Power*.
2. The crew reenergizes DC Bus 4 from Battery 4.
3. The US directs a PEO to place Charger 4 in service on DC Bus 4 per OP 3345C, *125 Volt DC*.
4. The PEO closes the Charger 4 DC Supply Breaker on Battery Bus 4.
5. The PEO then closes the Charger 4 DC Output Breaker to connect the Charger to the DC bus.

What is an impact of this operation described in OP 3345C, and what implication exists?

a) An in-rush of current may damage the Charger rectifier stack. If this occurs, the operators are required to open the Battery Output Breaker, and then energize the DC Bus from the Swing Charger.

b) An in-rush of current may damage the Charger rectifier stack. If this occurs, the operators are required to energize the DC Bus from the Swing Charger after closing its AC Input Breaker.

c) An in-rush of current may trip the DC output breaker. If this occurs, the operators are to wait at least 5 minutes, then cycle the DC Output breaker off and on.

d) An in-rush of current may trip the DC output breaker. If this occurs, the operators are to cycle the DC Output breaker off and on as promptly as possible.

Proposed Answer: D

Explanation (Optional): When returning a charger to service, the DC breakers connecting the charger to the bus are closed first, before the charger AC input breaker, in order to allow the DC battery bus to charge the charger filter capacitors, and if there is very little or no residual charge on the capacitors, the in-rush current may be high enough to cause one or more DC output breakers to trip. If this occurs, the operators are to as promptly as possible cycle the DC breakers, to tie the charger to the DC bus prior to the capacitors discharging again ("D" correct, "C" wrong). "A" and "B" are wrong, but plausible, since this is a misapplication of a Caution warning the operators of the potential to damage the Charger rectifier stack if the crew places a charger in service on a deenergized DC bus. "C" is plausible, since this is the correct impact, and as a general rule, operators are not to respond "as promptly as possible" to an unexpected system response.

Technical Reference(s):	AOP 3563 (Rev 10-1), Att. D, step 2	(Attach if not previously provided)
	OP 3345C (Rev 16-8), Precaution 3.6	
	OP 3345C (Rev 16-8), Note prior to step 4.16.4	
	OP 3345C (Rev 16-8), Section 4.16	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-03307 Describe the major administrative or procedural precautions and limitations associated with the 125 VDC Distribution System, including the basis for each…	(As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 14	Tier #	1	1
K/A Statement: Loss of Nuclear Service Water:	Group #	1	1
Reasons for actions contained in the EOP	K/A #	APE.062.AK3.03	
Proposed Question:	Importance Rating	4.0	4.2

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

- The crew has entered AOP 3560, *Loss of Service Water.*
- No "A" train Service Water (SWP) Pumps can be started.

Why does AOP 3560 direct the crew to start a second "B" train SWP Pump?

a) To ensure sufficient pressure exists in the "B" SWP Train to supply the MCC/Rod Control Area ACU.

b) To ensure sufficient pressure exists in the "B" SWP Train to supply the Control Building Chillers.

c) To ensure sufficient flow exists to supply two Reactor Plant Component Cooling Water (RPCCW) Heat Exchangers.

d) To ensure sufficient flow exists to supply two Turbine Plant Component Cooling Water (TPCCW) Heat Exchangers.

Proposed Answer: D

Explanation (Optional): "A" and "C" are wrong, since trains are not cross connected to protect the operable train by not placing the plant in an unanalyzed condition. "A" and "C" are plausible, since SWP is capable of being cross-connected at several locations including at the RPCCW Heat Exchangers. "B" is wrong, but plausible, since the MCC/RCA Booster Pump ensures adequate pressure exists at the MCC/RCA ACU, which is at a high elevation. "D" is correct, since the service water supply is required to two TPCCW heat exchangers to cool the secondary plant with the plant on line.

Technical Reference(s): AOP 3560 Basis Doc (Rev 8-1), step 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03928 Discuss the basis of major precautions, procedure steps, and/or step sequence (in AOP 3560). (As available)

Question Source: Bank 70389

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 15	Tier #	1	1
K/A Statement: Loss of Instrument Air:	Group #	1	1
Operate/monitor RPS	K/A #	APE.065.A1.05	
Proposed Question:	Importance Rating	3.3	3.3

With the plant at 100% power, a leak in the instrument air system occurs, and the following sequence of events occurs:

1400 The RO reports that instrument air pressure is decreasing at a moderate rate.
1401 The crew enters AOP 3562, *Loss of Instrument Air.*
1412 Letdown isolates
1414 PZR spray valves close
1415 Reactor Plant Chilled Water CTMT header isolates
1417 Feed Reg Valves close

What was the earliest time at which the operators were required by AOP 3562 to manually trip the reactor?

a) 1412

b) 1414

c) 1415

d) 1417

Proposed Answer: D

Explanation (Optional): The crew is directed to trip the reactor and go to E-0 when instrument air pressure is decreasing rapidly or when feedwater control is lost ("D" correct, and "A", "B", and "C" wrong). "A", "B", and "C" are plausible since they are plant responses that will occur on a loss of air that have adverse effects on the plant.

Technical Reference(s): AOP 3562 (Rev 7-1), step 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03941 Discuss conditions which require transition to other procedures (As available)

Question Source: Bank 76194

Question History:

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 # 16	Tier #	1	1
K/A Statement: LOCA Outside Containment:	Group #	1	1
Operational implications of emergency systems	K/A #	W/E04.EK1.1	
Proposed Question:	Importance Rating	3.5	3.9

A LOCA outside Containment occurs, and the following sequence of events occurs:

1. The crew enters E-0, *Reactor Trip or Safety Injection*.
2. At the E-0 step 16 brief, the RO reports the following parameters:
   - RCS pressure: 1650 psia and decreasing.
   - RHR flow: 300 gpm per pump.
   - SIH flow: 0 gpm per pump.
3. The crew enters ECA-1.2, *LOCA outside Containment.*
4. The US directs the RO to close Cold Leg Injection Valve 3SIL*MV8809A.

Immediately after closing 3SIL*MV8809A, the RO reports the following:

- RCS pressure: 1400 psia, and monitoring to determine a trend.
- RHR flow: 0 gpm per pump.
- SIH flow: 100 gpm per pump.

Correctly complete the following statement regarding the LOCA:

The RCS LOCA was originally into the___(1)____system, and the LOCA___(2)____been isolated from the RCS.

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

Proposed Answer: A

Explanation (Optional): Since RHR flow initially existed with RCS above RHR pump shutoff head, the leak was into the RHR System. Since RHR flow dropped to zero with the discharge valve closed, the leak is on the RCS side of the isolation valve, so the leak is still active ("C" and "D" wrong). The leak is not into SIH since RHR flow went to zero when the SIL valve was closed ("A" correct, "B" wrong). Note that isolating RHR will not seat the check valve, so leak remains active. "B" is plausible, since RHR flow dropped to zero when the SIL valve was closed. "C" and "D" are plausible, since SIH flow has also changed, but this was due to RCS pressure dropping below SIH pump shutoff head (1550 psia).

Technical Reference(s): P&ID 112A (Rev 50) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05459 Given a failure, partial or complete, of the Residual Heat Removal System, determine the effects on the system and on interrelated systems (As available)

Question Source: Bank 75467

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 17	Tier #	1	1
K/A Statement: Loss of Secondary Heat Sink:	Group #	1	1
Knowledge of the specific bases for EOPs.	K/A #	GEN.2.4.18	
	Importance Rating	3.3	4.0

Proposed Question:

Initial Conditions:

- The crew is at E-0, *Reactor Trip or Safety Injection*, step 17, "Verify Adequate Heat Sink."
- No AFW Pumps are running, and none can be started.

The crew is preparing to transition to FR-H.1, *Loss of Secondary Heat Sink*, and current parameters and trends are as follows:

- RCS pressure: 800 psia and slowly decreasing.
- SG pressures: 900 psig and slowly decreasing.
- CTMT temperature: 160°F and slowly increasing.
- SG levels:
  - "A" SG: 33% Wide Range (0% NR), and slowly decreasing
  - "B" SG: 35% Wide Range (0% NR), and stable
  - "C" SG: 35% Wide Range (0% NR), and stable
  - "D" SG: 27% Wide Range (0% NR), and slowly decreasing

What strategy will FR-H.1 direct, and why?

a) Return to E-0, since all SG pressures are decreasing.

b) Return to E-0, since RCS pressure is less than Steam Generator pressure.

c) Attempt to restore a heat sink from the Main Feed System, since the AFW System is unavailable.

d) Attempt to restore a heat sink by establishing bleed and feed of the RCS, since core cooling is in jeopardy.

Proposed Answer: B

Explanation (Optional): "B" is correct, and "A", "C", and "D" wrong, since with RCS pressure less than SG pressures, break flow is removing all decay heat and SGs are not required for heat sink. "A" is plausible, since SG pressure stable or decreasing is one of the criteria used to verify adequate natural circulation in the EOP network. "C" is plausible, since this is first method attempted in FR-H.1 with no AFW pumps available to restore heat sink. "D" is plausible, since CTMT temperature is elevated, and bleed and feed criteria for adverse CTMT is met (assuming RCS pressure is above SG pressure).

Technical Reference(s): FR-H.1 (Rev 21-0), step 1 (Attach if not previously provided)

FR-H.1 (Rev 21-0), Caution prior to step 3

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04536 Discuss conditions requiring transition to other procedures from EOP 35 FR-H.1... (As available)

Question Source: Modified Bank 78378 (Parent Question attached)

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10, 41.14, and 43.5

Comments:

Original Bank Question 78378 (prior to modification):

With the plant initially at 100% power, a large break LOCA occurs, and current conditions are as follows:

- The crew is performing EOP-FR-H.1, *Response to Loss of Secondary Heat Sink*, Step 1 "Check if Secondary Heat Sink is Required."
- The RCS had depressurized to CTMT pressure.

What action, if any, will FR-H.1 direct the crew to take concerning heat sink, and why?

a) FR-H.1 will not direct any actions to be taken to restore secondary heat sink, since a secondary heat sink is NOT required.

b) FR-H.1 will direct the crew to attempt to restore feed in order to establish a heat sink and minimize core uncovery.

c) FR-H.1 will direct the crew to attempt to restore feed in order to avoid SG dryout, minimizing thermal stresses on the affected S/G tubes.

d) FR-H.1 will direct the crew to attempt to restore feed in order to establish a thermal layer above the SG tubes, minimizing rad release should a SG tube rupture occur.

Correct answer: A

Considered "Modified" since original question was a large break LOCA, and modified question is small break LOCA. Also, modified question includes individual SG levels and CTMT temperature. Distractors have been modified to include returning to E-0 versus remaining in FR-H.1, and method of restoring heat sink in FR-H.1

Examination Outline Cross-reference:	Level	RO	SRO
Question # 18	Tier #	1	1
K/A Statement: Generator Voltage and Electric Grid	Group #	1	1
Disturbances: Determine/interpret Generator	K/A #	077.A2.03	
current outside the capability curve	Importance Rating	3.5	3.6

Proposed Question:

A partial loss of Main Generator hydrogen pressure has occurred, resulting in the following Initial Conditions:

- Real Load: 1225 MWe
- Reactive Load: 100 MVARs Out
- Generator Hydrogen Pressure: 60 psig

The Grid becomes unstable, and the BOP operator reports the following:

- Real Load: 1225 MWe
- Reactive Load: 300 MVARs Out
- Generator Hydrogen Pressure: 60 psig

Using OP3324A, Attachment 1, attached to this exam, is the Main Generator operating inside or outside the Turbine Generator Capability Curve; and what Main Generator parameter is most challenged by this transient?

a) The Main Generator is operating INSIDE the curve; and Main Generator armature heating is most challenged.

b) The Main Generator is operating INSIDE the curve; and Main Generator field heating is most challenged.

c) The Main Generator is operating OUTSIDE the curve; and Main Generator armature heating is most challenged.

d) The Main Generator is operating OUTSIDE the curve; and Main Generator field heating is most challenged.

Proposed Answer: C

Explanation (Optional): The transient has resulted in real load staying constant (grid frequency has not changed) and reactive load increasing (grid voltage has decreased). "A" and "B" are wrong, since parameters are now outside the Generator Capability Curve. "A" and "B" are plausible, since the new combination of MWe and MVAR is close to the curve, and several variables input to whether the point is inside or outside the curve, and if hydrogen pressure were normal, parameters would still be inside the curve. "C" is correct, and "D" wrong, since the combination of MWe and MVAR has placed the Generator outside of the BC curve, which is limited by armature heating. "D" is plausible, since if MWe were lower, and MVAR higher, the AB curve would be challenged, and this is limited by field heating.

Technical Reference(s): OP 3324A (Rev 10-6), Attachment 1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: **OP 3324A, Attachment 1**

Learning Objective: MC-04685 Describe operation of Main Generator… System under… Abnormal Voltage Operations… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4 and 41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 19	Tier #	1	1
K/A Statement: Loss of Condenser Vacuum:	Group #	2	2
Knowledge of specific bases for EOPs	K/A #	APE.051.GEN.2.4.18	
Proposed Question:	Importance Rating	3.3	4.0

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

0710: Condenser backpressure is 4.0 inches Hg abs and degrading rapidly
0711: The crew enters AOP 3559, *Loss of Condenser Vacuum*
0715: Condenser backpressure is 5.1 inches Hg abs and degrading. The crew begins a rapid downpower.
0730: Turbine load has been reduced to 380 MWe with Condenser backpressure at 7 inches Hg absolute and degrading.
0735: Turbine load has been reduced to 320 MWe with Condenser backpressure stable at 7.6 inches Hg absolute.

Correctly complete the following statement regarding the requirement to trip the reactor, and the associated basis.

In accordance with the foldout page of AOP 3559, the first time the crew was required to trip the reactor was at___(1)____, and the basis for tripping the reactor was to prevent___(2)____.

	(1)	(2)
a.)	0730	rupturing the LP Turbine diaphragms
b.)	0730	damaging the LP Turbine blades
c.)	0735	rupturing the LP Turbine diaphragms
d.)	0735	damaging the LP Turbine blades

Proposed Answer: B

Explanation (Optional): AOP 3559 foldout page criteria require a reactor trip above 3% power if either: condenser backpressure is greater than 7.5 in. Hg abs ("A" and "B" plausible) OR if condenser backpressure is greater than 5 inches Hg absolute AND the turbine is in service and loaded to 389 MWe or LESS. This is met at 0730. The basis for turbine operating limitations below 389 MWe with backpressure > 5 inches is to prevent LP turbine blade stall flutter, which can damage the LP Turbine Blades ("C" correct, and "D" wrong). "D" is plausible as LP turbine diaphragms are installed on all LP turbines. This event would raise backpressure; however, the event does NOT raise pressure enough (5 psig) to challenge the LP turbine diaphragms. Differentiating between the two proposed failure mechanisms given in this question is significant since one damages the turbine, and the other is a personnel safety concern that would also restrict access to the turbine building.

Technical Reference(s):	AOP 3559 (Rev 10-0), Foldout Page OP 3323A (Rev 15-5), Precaution 3.8	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-03923 Discuss the basis of major precautions, procedure steps/or sequence of steps contained within AOP 3559.	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.10, and 41.14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 20	Tier #	1	1
K/A Statement: High Reactor Coolant Activity:	Group #	2	2
Determine/interpret corrective actions	K/A #	APE.076.AA2.02	
Proposed Question:	Importance Rating	2.8	3.4

With the plant at 100% power, RCS activity starts increasing.

The crew enters AOP 3553, *High RCS Activity.*

Correctly complete the following statement regarding actions directed by AOP 3553 based on current plant conditions.

Consider restricting access to the___(1)____, and consult with___(2)____.

	(1)	(2)
a)	Auxiliary Building	Reactor Engineering about increasing letdown flow
b)	Auxiliary Building	Health Physics about placing SLCRS in service
c)	ESF Building	Reactor Engineering about increasing letdown flow
d)	ESF Building	Health Physics about placing SLCRS in service

Proposed Answer: A

Explanation (Optional): Reactor coolant with increasing activity levels is circulating through the Aux Bldg, resulting in increasing radiation levels in the Auxiliary Bldg. "C" and "D" are wrong, since Reactor Coolant is not flowing through the ESF Building at 100% power. "C" and "D" are plausible, since this action is required in lower MODES when RHR is in service, and ESF Bldg radiation levels are also an issue during a LOCA when the RCS is on sump recirculation. AOP 3553 step 7 directs the operators to consult with RE about increasing letdown flow ("A" correct) to increase flow through the demins and filters to enhance purification of Reactor Coolant. "B" is wrong, since SLCRS is effective at removing airborne activity, but not higher activity levels in the charging and letdown piping. "B" is plausible, since SLCRS draws on both the ESF and Aux Bldg, and if an RCS leak occurs in either building, increasing radiation levels in either building will drive starting the SLCRS system per AOP 3573, *Radiation Monitor Alarm Response.*

Technical Reference(s): AOP 3553 (Rev 6-4), Note prior to step 1 (Attach if not previously provided)
AOP 3553 (Rev 6-4), step 7

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07530 Given a set of plant conditions, properly apply the notes and cautions of AOP 3553. (As available)

Question Source: Bank 76174

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10 and 41.12

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 21	Tier #	1	1
K/A Statement: Steam Generator Over-pressure	Group #	2	2
Knowledge of parameters and logic used to	K/A #	EPE.W/E13.GEN. 2.4.21	
assess the status of safety functions	Importance Rating	4.0	4.6

Proposed Question:

Initial Conditions:

- The crew is preparing to transition out of E-0, *Reactor Trip or Safety Injection*.
- Manual Status Tree monitoring is in effect.

The BOP reports the following parameters for the HEAT SINK Status Tree.

- Total AFW Flow: 450 gpm
- "A" SG pressure: 1230 psig
- "B" SG pressure: 1125 psig
- "C" SG pressure: 1125 psig
- "D" SG pressure: 1100 psig
- "A" SG NR level: 75%
- "B" SG NR level: 20%
- "C" SG NR level: 25%
- "D" SG NR level: 10%
- CTMT temperature: 120°F

What is the status of the HEAT SINK Status Tree, and why?

a) RED, due to inadequate heat sink

b) YELLOW, due to SG overpressure

c) YELLOW, due to SG high level

d) YELLOW, due to SG low level

Proposed Answer: B

Explanation (Optional): This is considered acceptable for the RO portion of the exam, since RO objectives exist for yellow path entry conditions, and minimal assessment of plant conditions exists in this question, just recognizing which entry condition is met. The parameters and logic used to assess the status of the Heat Sink safety function are as follows: Heat Sink Red is ALL SG NR Levels <8% AND AFW flow <530 gpm ("A" wrong, but plausible). Yellow on SG overpressure is ANY SG >1220 psig ("B" correct). Yellow on SG high level is ANY SG NR >80% ("C" wrong, but plausible). Yellow on SG low level is ANY SG NR <8% ("D" wrong, but plausible).

Technical Reference(s): EOP 35 F-0.3 (Rev 5-1) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05955 Identify plant conditions that require entry into EOP 35 FR-H.2 (As available)

Question Source: New

Question History:

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 # 22	Tier #	1	1
K/A Statement: Post LOCA Cooldown Depress:	Group #	2	2
Operational implications of procedures	K/A #	W/E03.EK1.2	
Proposed Question:	Importance Rating	3.6	4.1

A fault occurs in the "A" SG, resulting in the following sequence of events:

1. The crew enters E-1, *Loss of Reactor or Secondary Coolant.*
2. The crew reaches E-1, step 9, "Check RCS and SG Pressures."
3. Since the "A" SG pressure is still decreasing, the crew is directed to return to E-1, step 1.

Why is it important that the operators NOT proceed past E-1, Step 9 with the "A" SG depressurizing?

a) The crew would be directed to transition to ES-1.2, *Post LOCA Cooldown and Depressurization*, where unnecessarily restrictive SI Termination Criteria would be encountered.

b) E-1 provides no guidance for faulted steam generator isolation past this point.

c) The RCS cooldown rate must be under operator control in order for subsequent E-1 steps to be effectively implemented.

d) The crew would be directed to transition to ES-1.3, *Transfer to Cold Leg Recirculation*, and aligning for cold and hot leg recirculation is not desired for a steam line break.

Proposed Answer: A

Explanation (Optional): "A" is correct, since if the operators proceed past Step 9 in E-1 with a depressurizing SG, they will be directed to ES-1.2, *Post LOCA Cooldown And Depressurization* ("D" wrong), and encounter more restrictive SI termination criteria than necessary, since ES-1.2 is designed to cooldown the plant and sequentially terminate ECCS injection with a small break LOCA in progress. For a faulted SG, RCS inventory will rapidly recover after the faulted SG has completed blowing down. By looping back at E-1, step 9, the crew will transition to ES-1.1, *SI Termination*, and terminate ECCS injection much more quickly. "B" and "C" are wrong, since operators are not directed to start a cooldown in E-1, and the faulted SG has already been addressed in E-2, *Faulted Steam Generator Isolation*. "B", "C" and "D" are plausible, since E-1 will transition the crew to the appropriate procedure with a faulted SG, a small break LOCA, or a large break LOCA. "C" is plausible, since ES-1.2 will conduct a cooldown of the RCS. "D" is plausible, since a transition to ES-1.3 exists after step 9.

Technical Reference(s): E-1 (Rev 26-0), step 9; WOG Bkgd Doc (Rev 2), E-1, Step 9 Basis (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04362 Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 E-1 (As available)

Question Source: Bank 70254

Question History:

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 # 23	Tier #	1	1
K/A Statement: Turbine Trip: Operational	Group #	2	2
implications of systems or procedures	K/A #	AOP 3550.AK1.01	
Proposed Question:	Importance Rating	Site Priority	Site Priority

With the plant initially at 30% power, the following sequence of events occurs:

1. The Main Turbine trips.
2. The crew enters AOP 3550, *Turbine/Generator Trip.*
3. The Reactor does NOT trip during the initial transient.

What action will AOP 3550 direct the crew to take with Control Rods, and why?

a) Verify the Control Rods are in AUTO, so they are available to automatically insert to restore Tave to program.

b) Verify the Control Rods are in AUTO, so they are available to automatically insert in the event of a Condenser Steam Dump malfunction.

c) Place the Control Rods in MANUAL when reactor power decreases to 20 to 25%, to stabilize reactor power and obtain feedwater control on the SG Feed Reg Bypass Valves.

d) Place the Control Rods in MANUAL when reactor power decreases to 20 to 25%, to allow Condenser Steam Dumps to restore Tave to program in a controlled manner.

Proposed Answer: C

Explanation (Optional) On a turbine trip, a C-7 signal (Load Reject) is likely to actuate. This arms and opens condenser steam dump valves. The steam dumps have a 2°F deadband, and rods have a 1.5°F deadband. If both systems are left in auto, rods will continue to step in to lower temperature to within 1.5°F of program, closing the steam dumps and reducing power to below the point where Main Feedwater Control Valves can control SG levels. Because of this, AOP 3550, step 2 will direct the crew to place rods in manual when power is between 20 and 25% ("A" and "B" wrong). The purpose of this step is to stabilize reactor power in the 20% to 25% power range in order to obtain feedwater control on the SG feed regulating bypass valves ("C" correct). The rod control system must be placed in manual operation to stabilize reactor power in order to provide for turbine recovery or a controlled reactor shutdown. Without these actions, the rod control system would automatically insert the control rods to; 1), return Tavg to the no-load Tave ("D" wrong, "A" plausible), and 2), to close the condenser steam dump valves which would be open due to the temperature mismatch generated by the proportional only Tavg Mode steam dump controller ("B" plausible). "D" is plausible, since this is what would happen if rods are left in auto.

Technical Reference(s): AOP 3550 Basis Doc (Rev 8-0), step 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03897 Discuss the basis of major procedure steps and/or sequence of steps in AOP 3550. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1, 1, 6, 7, and 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 24	Tier #	1	1
K/A Statement: Severe Weather:	Group #	2	2
Determine/interpret conditions or procedures	K/A #	AOP 3569.A2.01	
Proposed Question:	Importance Rating	Site Priority	Site Priority

Current Conditions:

- A hurricane warning has been issued for southeastern Connecticut.
- Winds are projected to exceed 90 mph in the next 6 hours.
- CONVEX has reported that offsite power is NOT reliable.
- A plant shutdown is being performed as required by AOP 3569, *Severe Weather Conditions*.

What final plant conditions are required to be established by the crew prior to the hurricane arriving?

a) RCS T_{cold} at 557°F, and Pressurizer level at 28%

b) RCS T_{cold} at 557°F, and Pressurizer level at 60%

c) RCS T_{cold} less than 400°F, and Pressurizer level at 28%

d) RCS T_{cold} less than 400°F, and Pressurizer level at 60%

Proposed Answer: D

Explanation (Optional): This is acceptable for the RO portion of the exam , since RO objectives exist to know procedure notes, cautions, and foldout page items. This is a big-picture strategy question rather than a question requiring assessment of plant conditions. "D" is correct since with expected winds greater than 90 mph or offsite power not reliable or with either EDG inoperable, Pressurizer level will be increased to approximately 60% to provide an additional margin to core uncovery ("A" and "C" wrong). A cooldown will be conducted to cool the RCS to less than 400°F to provide a stable condition consistent with ECA-0.0, Loss of All AC Power., to protect the RCP seals. ("D" correct and "B" wrong), maintaining RCS pressure GREATER THAN 850 psia so that the SI accumulators do not inject. The accumulator isolation valves are kept open so that the accumulators are available in the event of a loss of all AC power. "A", "B", and "C" are plausible, since 557°F is normal HOT STANDBY program temperature and 28% Pzr level is normal HOT STANDBY program Pzr level; and this action is directed in AOP 3569 if flooding is anticipated at the intake structure. Also, numerous Technical Specification Action Statements require going to HOT STANDBY within 6 hours.

Technical Reference(s): AOP 3569 (Rev 18-0), steps 11, 12 and 14 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06216 Identify conditions which require transition to other procedures from AOP 3569, Severe Weather Conditions (As available)

Question Source: Bank 64019

Question History: Millstone 3 2002 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 # 25	Tier #	1	1
K/A Statement: Rapid Downpower:	Group #	2	2
Operate/monitor indications, plant behavior,	K/A #	AOP 3575.A1.01	
and/or desired results	Importance Rating	Site Priority	Site Priority

Proposed Question:

The crew is performing a CONVEX requested emergency generation reduction from 100% power to 80% power, and the following sequence of events occurs:

1. During the load reduction axial flux difference goes out of the target band in the negative direction.
2. At 85% power, a PEO is dispatched to degrade condenser vacuum.
3. The Rod Bank LO-LO setpoint is reached.
4. The crew reaches the desired 80% power level.

Did AOP 3575 require the crew to stop the downpower prior to them reaching 80% power? If so, why?

a) The crew was required to temporarily stop the downpower when AFD went outside the target band. After borating to restore AFD, they could recommence the load reduction.

b) The crew was required to stop the downpower when the Rod LO-LO annunciator was received, and transition to AOP 3566, *Immediate Boration.*

c) The crew was required to stop the downpower at 85% power while waiting for condenser vacuum to degrade to minimize the chance of turbine rubbing.

d) The crew was NOT required to stop the downpower prior to reaching 80% power for the AFD, Rod LO-LO, or turbine rubbing concerns.

Proposed Answer: D

Explanation (Optional): "D" is correct, and "A", "B", and "C" wrong, since none of the three conditions requires the downpower to be stopped. "A" is plausible, since continuing the downpower will cause rods to continue to insert, aggravating the condition, but the downpower is important for grid stability, so the crew will be directed to increase the boration rate to minimize rod insertion while the downpower continues. "B" is plausible, since rod lo-lo requires immediate boration, but this is accomplished in AOP 3575 while the downpower is allowed to continue. "C" is plausible, since turbine rubbing is a concern during a rapid downpower, but only after lower power levels (<75%) are reached.

Technical Reference(s): AOP 3575 (Rev 18-0), Note prior to step 1 (Attach if not previously provided)
AOP 3575 (Rev 18-0), steps 6.h, 7. a-d, and 9

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07570 Given a set of plant conditions, properly apply the notes and cautions of AOP 3575. (As available)

Question Source: Bank 68689

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 26	Tier #	1	1
K/A Statement: Loss of Emergency Bus:	Group #	2	2
Operational implications of systems or	K/A #	AOP 3577.K1.01	
procedures and/or desired results	Importance Rating	Site Priority	Site Priority

Proposed Question:

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

1. The BUS 34C BUS DIFF annunciator is received on MB8A.
2. The crew enters AOP 3577, *Loss of Normal and Offsite Power to a 4.16 KV Emergency Bus.*

Which of the following actions is **NOT** procedurally directed for the loss of Bus 34C?

a) Locally shift the RCP seal return path to the top of the VCT.

b) Simultaneously isolate Charging and Letdown.

c) Transfer Non-Vital Instrument Bus 6 to the Alternate AC Source.

d) Isolate Auxiliary Steam to the Auxiliary Building.

Proposed Answer: A

Explanation (Optional): "A" is correct, since the seal water heat exchanger is cooled from the "B" RPCCW train, and shifting seal return to the top of the VCT allows mixing of hot seal return water prior to delivery into the Charging Pump suctions and to the RCP seals. "B" is wrong since the "A" RPCCW train cools the letdown heat exchanger, and letdown needs to be isolated to prevent VCT heatup. "C" is wrong, since on a loss of Bus 34C, the normal supply for IAC 6 (MCC32-3T) has been lost, and after 30 minutes, the backup DC supply will be lost, causing a loss of the Plant Process Computer if IAC 6 is not transferred to the alternate source. "D" is wrong, since cooling has been lost to the "A" RPCCW non-safety header, and relief valves may lift on equipment supplied by Aux Steam. "B" and "C" are plausible, since these actions are taken in AOP 3577 for loss of the "A" train. "D" is plausible, since this action is taken on loss of either Train.

Technical Reference(s): AOP 3577 (Rev 1-4), Steps 1, 8, 9, and 15 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07395 Describe the major action categories contained within AOP-3577, Loss Of Normal and Offsite Power to a 4.16kv Emergency Bus. (As available)

Question Source: Bank 78907

Question History:

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 # 27	Tier #	1	1
K/A Statement: Loss of All AC Power – Recovery with the SBO Diesel: Interrelations with Control, Safety, and/or Heat Removal Systems	Group #	2	2
	K/A #	ECA-0.3.K2.01	
	Importance Rating	Site Priority	Site Priority

Proposed Question:

Initial Conditions:

- A plant heatup is in progress per OP 3201, *Plant Heatup* after a three day mid-cycle shutdown.
- RCS temperature is 420°F.
- RCS pressure is 750 psia.

The following sequence of events occurs:

1. A loss of all AC Power occurs.
2. The crew restores power to the "A" Train with the SBO Diesel.
3. The crew enters ECA-0.3, *Loss of all AC Power – Recovery with the SBO Diesel.*
4. The crew starts the "A" RPCCW Pump.

Which other loads are assumed to be started per ECA-0.3 in order to meet the 8-hour SBO coping requirement?

a) A Service Water Pump and a Charging Pump

b) A Service Water Pump and an RHR Pump

c) The Emergency Seal Oil Pump and a Charging Pump

d) The Emergency Seal Oil Pump and an RHR Pump

Proposed Answer: A

Explanation (Optional): Considered RO level since the question can be answered with an understanding of the major strategy of ECA-0.3. The SBO diesel is not rated to support full loading of an emergency bus, so ECA-0.3 prioritizes equipment loading. "A" is correct, since Service Water is required to support Charging Pump cooling, and the Charging pump ensures the 8-hour coping duration is met. "B" and "D" are wrong, since the RHR Pump is not needed during normal operations until MODE 4, and the 8 hour coping time does not require a plant cooldown. Also, no LOCA is assumed, so RHR is not needed for RCS Injection. If injection is needed, ECA-0.3 will increase flow from the running Charging pump by opening a Cold Leg Injection Valve. "C" is wrong, since the Emergency Seal Oil Pump is DC-powered from Battery 6, and depending on the train the SBO diesel is placed, it may not regain power via the SBO diesel. "B" and "D" are plausible, since the RHR pumps are safety related pumps involved with maintaining either RCS inventory or heat removal. "C" is plausible, since ECA-0.0 takes action to extend battery life, but the assumed life of vital batteries is 2 hours, not 8 hours.

Technical Reference(s): ECA-0.3 (Rev 13-2), steps 5, 6, and 7 (Attach if not previously provided)
SFRM (Rev 6) page 4.2-7
SFRM (Rev 6 CNN 5) Section 2.17.6.4

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03866 Describe the major action categories within EOP35 ECA-0.3 (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 28	Tier #	2	2
K/A Statement: Reactor Coolant Pump:	Group #	1	1
Design feature/interlock which provides for:	K/A #	003.K4.11	
Isolation valve interlocks	Importance Rating	3.0	3.0

Proposed Question:

Which of the following combinations of RCS Isolation Valve positions will meet the electrical interlock for starting a RCP?

a) Cold Leg - CLOSED
Hot Leg - CLOSED
Bypass - OPEN

b) Cold Leg - CLOSED
Hot Leg - OPEN
Bypass - CLOSED

c) Cold Leg - OPEN
Hot Leg - CLOSED
Bypass - OPEN

d) Cold Leg - OPEN
Hot Leg - CLOSED
Bypass - CLOSED

Proposed Answer: A

Explanation (Optional): Either of two sets of valve positions are required to meet the interlock to allow starting an RCP: 1) The Hot and Cold leg isolation valves OPEN (None of the given alignments meets this portion of the interlock), or 2) The Cold leg isolation valve CLOSED ("C" and "D" wrong) with the Bypass Valve OPEN ("A" correct, and "B" wrong). "B", "C", and "D" are plausible, since they each have at least one valve open and involve the three valves that feed into the interlock.

Technical Reference(s): LSK 25-1.1A (Rev 8) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-00148 Describe the purpose and operation of the controls and interlocks associated with the operation of the Reactor Coolant Pumps. (As available)

Question Source: Bank 67470

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.3 and 41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 29	Tier #	2	2
K/A Statement: Chemical and Volume Control	Group #	1	1
Effect of a malfunction on: PZR level and pressure	K/A #	004.K3.07	
Proposed Question:	Importance Rating	3.8	4.1

With the plant at 100% power, an 8 gpm leak starts downstream of Charging Flow Control Valve 3CHS*FCV121.

Assuming no operator action, what indications will exist thirty minutes after the leak started?

a) VCT level constant and pressurizer level decreasing.

b) Increased VCT makeup and pressurizer level restored to normal.

c) Increased VCT makeup and pressurizer level decreasing.

d) Charging pump suction swapped to the RWST and PZR level restored to normal.

Proposed Answer: B

Explanation (Optional): A leak downstream of CHS*FCV121 would INITIALLY result in a decrease in pressurizer level. FCV 121 will open to restore PZR level and then maintain level on program, since the leak is within the capacity of the charging system, and the automatic level controller is a PID controller ("A" and "C" wrong). Because charging is now greater than letdown, VCT level will decrease to the makeup setpoint, and 8 gpm is within the capacity of VCT makeup ("B" correct and "D" wrong). "A" is plausible, since this is how the system would respond in manual. "C" and "D" are plausible, since a leak in the Charging System will impact the amount of water reaching the RCS, and VCT level will also be impacted as the Charging System responds.

Technical Reference(s): Functional Dwg 11 (Rev H) (Attach if not previously provided)
Process Control Sh. 27 (Rev J)
P&ID 104A (Rev 54)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04202 Describe the operation of the Chemical and Volume Control System under normal, abnormal, and emergency operating conditions. (As available)

Question Source: Bank 68571

Question History:

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: Residual Heat Removal:	Group #	1	1
Effect of a malfunction on: RCS	K/A #	005.K3.01	
Proposed Question:	Importance Rating	3.9	4.0

The plant is initially in MODE 5, preparing to return the plant to power at the end of a refueling outage. Initial conditions are as follows:

- The Pressurizer is solid.
- The "A" RHR train is in service.
- RCS Temperature is stable at 145^{o}F.

A large instrument air header rupture occurs, and instrument air pressure rapidly depressurizes to zero psig.

Assuming no operator action is taken, complete the following statement about RCS temperature response:

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

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

Proposed Answer: D

Explanation (Optional): This question matches the KA requiring the candidate to determine the effect of a RHR malfunction on the RCS, since the loss of air causes the RHR system to malfunction, and the candidate must then determine the effect of the RHR malfunction on the RCS. The plant is post-refueling, so decay heat is at a minimum, meaning 3RHS* FCV606 is initially throttled mostly closed. A loss of IAS will cause 3CCP*FV66A to fail AS IS resulting in NO change in RCS temperature from CCP flow ("A" and "C" wrong). "D" is correct, and "B" wrong, since RHR Flow Control Valve 3RHS*HCV 606 fails open on a loss of IAS, resulting in maximum flow through the RHR HX. "A", "B", and "C" are plausible, since temperature varies in the appropriate direction based on the assumed fail position in the distractor.

Technical Reference(s): AOP 3562 (Rev 7-1), page 3 (Attach if not previously provided)
P&ID 112A (Rev 50)
P&ID 121A (Rev 32), and 121C (Rev 36)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05459 Given a failure, partial or complete, of the Residual Heat Removal system determine the effects on the system and on interrelated systems (As available)

Question Source: Bank 73098

Question History:

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 # 31	Tier #	2	2
K/A Statement: Residual Heat Removal:	Group #	1	1
Predict impact and mitigate: RHR valve	K/A #	005.A2.04	
malfunction	Importance Rating	2.9	2.9

Proposed Question:

Initial Conditions:

- RCS temperature is 150°F.
- RCS Pressure is 150 psia.
- No RCPs are running.
- The "B" Train outage is in progress.

The following sequence of events occurs:

1. "A" RHR Pump Suction Valve 3RHS*MV8701A spuriously CLOSES.
2. The crew enters EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory.*

In accordance with EOP 3505, what is the crew required to do with RCS pressure?

a) Raise RCS pressure to greater than 170 psia, to ensure subcooled natural circulation occurs.

b) Vent the RCS to the PRT, to establish bleed and feed cooling of the RCS.

c) Depressurize the RCS to atmospheric to prevent lifting COPPS relief valves as the natural circulation ΔT develops.

d) Depressurize the RCS to atmospheric to prevent a cold overpressure event as makeup is added to the RCS.

Proposed Answer: A

Explanation (Optional): The "A" RHR Pump has lost suction, and must be tripped. The RCS is already full and steam generators are available. The procedure establishes conditions for natural circulation. This includes increasing RCS pressure to support the heatup and the steam generators are used to dump steam. "A" is correct, since natural circulation will proceed when RCS temperature increases to approximately 50°F greater than the saturation temperature of the secondary water. The required RCS pressure to maintain the RCS subcooled at the lowest pressure point in the RCS (SG U-Tubes), including instrument uncertainties, is 170 psia. "B" is wrong, since the 170 psia requirement is part of establishment of conditions for natural circulation. "C" and "D" are wrong, since the RO will be maintaining RCS pressure between 170 and 330 psia. "B" is plausible, since there is a pressure requirement for RCS pressure in EOP 3505 when running an RCP, but the pressure band is 310 to 375 psia. "C" is plausible, since the PZR may be solid when in MODE 5, and a heatup will cause an increase in RCS pressure. "D" is plausible, since this is a misapplication of the PTS caution that applies when adding makeup via a high head source.

Technical Reference(s):	EOP 3505 (Rev 11-0), Att. B, steps 11 and 12 EOP 3505 (Rev 11-0), Att. B, Caution prior to step 1 OP 3260A (Rev 17-6), step 1.3.2	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04352 Discuss the bases of major procedure steps and/or sequence of steps in EOP 3505, Loss of Shutdown Cooling and/or RCS Inventory	(As available)

Question Source: Bank 80858

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10 and 41.14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 32	Tier #	2	2
K/A Statement: Emergency Core Cooling:	Group #	1	1
Physical connections and/or cause-effect	K/A #	006.K1.09	
relationship with: Nitrogen	Importance Rating	2.6	2.9

Proposed Question:

With the plant at 100% power, the RO is directed to raise pressure in the "D" SIL Accumulator using OP 3310B, *Low Pressure Safety Injection.*

Correctly complete the following statement regarding required valve operation and regarding the concern with pressurizing the nitrogen header for an extended period of time per OP 3310B.

___(1)___Nitrogen Supply Containment Isolation Valves (3SIL*CV8880 and 8968) prior to pressurizing the "D" Accumulator nitrogen supply line. Additionally, minimize the time the "D" Accumulator nitrogen header is pressurized due to concerns with___(2)___.

a) (1) VERIFY OPEN the normally open
(2) inerting the Containment atmosphere with nitrogen

b) (1) VERIFY OPEN the normally open
(2) over-pressurizing the "D" Accumulator

c) (1) OPEN the normally closed
(2) inerting the Containment atmosphere with nitrogen

d) (1) OPEN the normally closed
(2) over-pressurizing the "D" Accumulator

Proposed Answer: C

Explanation (Optional): The Accumulators are supplied by a normally-depressurized common nitrogen header. This header is normally isolated by two closed containment isolation valves (3SIL*CV8880 and 8968). "A" and "B" are wrong since the normal at power alignment of the containment isolation valves (CIV's) is OPEN. "B" and "D" are plausible since several CIV's for standby systems are maintained open at power. All four Accumulators vent to the Containment atmosphere through two parallel-path vent valves via a common vent header that taps off of the supply headers. "C" is correct, and "B" wrong, since the vent header taps off the pressurization header, and the vent valves leak by to the Containment atmosphere. A NOTE warns of this concern in OP 3310B "Do not leave accumulator nitrogen supply header pressurized for long periods of time. Leakage by accumulator vent control valves inerts containment atmosphere." "A" and "D" are plausible, since nitrogen is supplied from a high pressure source. However, the associated procedure section locally verifies the supply pressure control valve (3SIL-PIC8893) is properly set to 660 psig and in "AUTO" prior to beginning the evolution.

Technical Reference(s): OP 3310B (Rev 15-6), Section 4.6 (Attach if not previously provided)

OP 3310B (Rev 15-6), Precautions 3.3, 3.5, and 3.6

P&ID 112B (Rev 23)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06288 Describe the major administrative or procedural precautions and limitations placed on the operation of the Emergency Core Cooling System, and the basis for each. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 33	Tier #	2	2
K/A Statement: Pressurizer Relief/Quench Tank:	Group #	1	1
Ability to manually operate and/or monitor:	K/A #	007.A4.10	
Recognition of leaky PORV/code safety	Importance Rating	3.6	3.8

Proposed Question:

With the plant at 100% power, a Pressurizer PORV starts leaking by, and the following indications exist:

- Pressurizer pressure: 2235 ~~psig~~ psia and DECREASING.
- Tave: 587°F.
- PRT pressure: 20 psia and INCREASING

R 12/9/13

The RO monitors PORV tail pipe temperature on MB4.

What will tail pipe temperature indicate?

a) 230°F

b) 300°F

c) 617°F

d) 650°F

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since the enthalpy of the saturated steam in the PZR vapor space does not change as it passes through a relief valve, resulting in a temperature indication corresponding to the pressure in the PRT. The Mollier diagram shows that enthalpy of the PZR steam is approximately 1120 BTU/lb. Move across (left to right) to the PRT pressure, demonstrating that the steam will be saturated at PRT pressure. Follow the constant pressure line up to the saturation curve, which reads about 230°F. "B" is plausible, since this is the temperature obtained if a constant entropy process is assumed. "C" and "D" are plausible, since these are the approximate temperatures of the RCS hot leg temperature for the current Tave, and the temperature of the Pressurizer, and would then be corrrect if a constant temperature process is assumed (TMI).

Technical Reference(s): Steam Tables, Mollier Diagram (Attach if not previously provided)

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

Learning Objective: MC-05349 Describe the Pressurizer Relief Tank System operation… under the following… Pressurizer Safety Valve or Power Operated Relief Valve discharge… (As available)

Question Source: Bank #75623

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3 and 41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 34	Tier #	2	2
K/A Statement: Component Cooling Water:	Group #	1	1
Predict and/or monitor parameters associated with	K/A #	008.A1.02	
operating controls including: CCW temperature	Importance Rating	2.9	3.1

Proposed Question:

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

- The plant is at 340°F.
- The "B" Train of RHR has just been aligned for plant cooldown.

The RO slowly throttles the "RPCCW HX FLOW" controller (3CCP-HK66B1) in the open direction.

What effect, if any, does the RO's action have on RPCCW System temperature; and what is the control system response, if any?

a) No effect. The control system did NOT respond, since full RPCCW flow was already flowing through the "B" RPCCW Heat Exchanger.

b) No effect. The control system did NOT respond, since full RPCCW flow was already flowing through the "B" RHR Heat Exchanger.

c) Temperature increases. When higher RPCCW temperature is detected at the "B" RPCCW Heat Exchanger outlet, RPCCW flow through the RPCCW Heat Exchanger will increase.

d) Temperature increases. When higher RPCCW temperature is detected at the "B" RHR Heat Exchanger outlet, RPCCW flow through the RPCCW Heat Exchanger will increase.

Proposed Answer: C

Explanation (Optional): At this point, RHR has been selected to the Cooldown Mode This restores instrument air to the RHR Heat Exchanger outlet flow control valve and bypass valve, allowing them to modulate ("A" and "B" wrong). "A" and "B" are plausible, since placing RHR into the Normal Mode isolates instrument air to the RHR Heat Exchanger outlet flow control valve and bypass valve, causing them to fail open. Also, if RPCCW temperature continues to rise as sensed at the RHR Heat Exchanger , the RHR Heat Exchanger Total Flow Control Valve 3RHS*FCV619 will fail open. "C" is correct, since as RHR System temperature increases, as sensed at the discharge of the "B" Train RPCCW Heat Exchanger ("D" wrong, but plausible), RPCCW Temperature Control Valve 3CCP*TV32B will modulate to provide more RPCCW flow through the RPCCW Heat Exchanger to help mitigate the temperature rise.

Technical Reference(s): OP 3208 (Rev 22-4), steps 4.3.15 and 16 (Attach if not previously provided)
P&ID 112A (Rev 50) and 121A (Rev 32)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04154 Describe the operation of the Reactor Plant Component Cooling System under the following... Plant Cooldown... (As available)

Question Source: New

Question History:

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 # 35	Tier #	2	2
K/A Statement: Component Cooling Water:	Group #	1	1
Ability to recognize abnormal operating parameters	K/A #	008.GEN.2.4.4	
that are entry conditions for EOPs and AOPs	Importance Rating	4.5	4.7

Proposed Question:

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

1. The crew enters AOP 3561, *Loss of Reactor Plant Component Cooling Water*.
2. The STA reports the following parameters:
   - RCP Seal Injection flow is 6.5 gpm per RCP and stable.
   - RCP A and D bearing oil temperatures are 190°F and increasing.
   - RCP A and D Number 1 seal inlet temperatures are 200°F and increasing.
   - VCT temperature is 140°F and stable.

At this point, based on AOP 3561 Foldout page criteria, which parameters requires the crew to trip the reactor and enter E-0, *Reactor Trip or Safety Injection*?

a) Seal Injection flow.

b) Bearing oil temperature.

c) Seal inlet temperature.

d) VCT temperature.

Proposed Answer: D

Explanation (Optional): The reactor must be tripped and E-0 entered at this power level if RCP trip criteria are met. . "A" is wrong, since low seal injection flow is not one of the foldout page criteria. "A" is plausible, since seal injection flow is low, and thermal barrier cooling has been lost to some of the RCPs. RCP Trip Criteria include RCP Bearing Oil temperature >195°F ("B" wrong, but plausible), Number 1seal inlet temperature ≥ 230°F ("C" wrong, but plausible). Above 400°F, all four RCPs are required to be tripped if VCT temperature is >135°F ("D" correct) but RCP trip criterion would not be met with the plant shutdown and cooled down to <400°F, where the trip criterion would be VCT temperature >150°F

Technical Reference(s): AOP 3561 (Rev 11-2), Foldout Page (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07542 Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of AOP 3561. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 36	Tier #	2	2
K/A Statement: Pressurizer Pressure Control:	Group #	1	1
Design feature/interlock which provides for:	K/A #	010.K4.03	
Over pressure control	Importance Rating	3.8	4.1

Proposed Question:

The plant is initially at 100% power with PZR pressure stable at 2250 psia.

Pressurizer control heater group 3RCS-HIC fails to the fully energized condition.

Assuming no operator action is taken, how will the Pressurizer Pressure Control System respond to the increase in RCS pressure?

a) RCS pressure will increase to the spray valve open setpoint of 2325 psia, and spray valve(s) will open to lower pressure to 2275 psia, where the spray valve(s) will close. Pressure will cycle between 2325 psia and 2275 psia.

b) RCS pressure will increase to the spray valve open setpoint of 2325 psia, and spray valve(s) will open to lower pressure back to 2250 psia.

c) RCS pressure will increase to the spray valve setpoint of 2275 psia, and spray valve(s) will throttle open to maintain pressure at 2275 psia.

d) RCS pressure will increase to the spray valve setpoint of 2275 psia, and spray valve(s) will throttle open to stabilize pressure, and then slowly continue to throttle open to lower pressure back to 2250 psia.

Proposed Answer: D

Explanation (Optional): "D" is correct, since the spray valves start to throttle open at 2275 psia ("A" and "B" wrong), and receive a full open signal at 2325 psia; and the controller is a PID controller, so the longer the error exists, the "I" portion will cause the output to continue to increase to lower pressure until pressure is restored to 2250 psia ("C" wrong). "A" is plausible, since these pressures relate to spray valve operations and this is how PORVs work to control pressure. "B" is plausible, since this has the "PI" function of spray valve operations. "C" is plausible since this is how the controller would work if it were a "P" controller.

Technical Reference(s): Functional Dwg 11 (Rev H) (Attach if not previously provided)
Process Dwg 26 (Rev J)
Training Drawing PPL010C-4 (Rev 0)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05338 Describe the operation of the Pressurizer Pressure and Level Control System Controls and Interlocks… Pressurizer Master Pressure Controller… (As available)

Question Source: Bank 80879

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 37	Tier #	2	2
K/A Statement: Reactor Protection:	Group #	1	1
Effect of a malfunction on: CRDS	K/A #	012.K3.01	
Proposed Question:	Importance Rating	3.9	4.0

The plant is operating at 100% power, with Reactor Trip Breaker testing in progress, and initial conditions are as follows:

- Reactor Trip Breakers (RTB) "A" and "B" are closed.
- Operators have just closed Reactor Trip Bypass Breaker (BYB) "A".

The following sequence of events occurs:

1. The "D" Main Feed Reg Valve fails closed, resulting in a valid automatic reactor trip signal.
2. The "B" Train of RPS fails to actuate.

Assuming the "A" Train of RPS operates as designed, and no operator action is taken, how do RTB "B" and BYB "A" initially respond?

	RTB "B"	BYB "A"
a)	Trips	Does NOT trip
b)	Does NOT trip.	Trips
c)	Does NOT trip.	Does NOT trip.
d)	Trips	Trips

Proposed Answer: C

Explanation (Optional): "C" is correct since the AUTO trip signal from "A" Train RPS sends shunt and UV trip signals to the "A" Trip Breaker and a UV trip to the "B" Bypass Breaker. "A" and "D" are wrong since Reactor Trip Breaker "B" does not receive an auto trip signal from the "A" train. "B" is wrong since Bypass Breaker "A" does not receive an auto trip signal from the "A" train. "A", "B", and "D" are plausible, since depending on the trip signal type (auto versus manual) and train, shunt and UV trips are sent to various combinations of all four breakers.

Technical Reference(s): Functional Dwg 2 (Rev N) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05493 Describe the operation of the following RPS controls and interlocks… Reactor Trip and Bypass Breakers… (As available)

Question Source: Bank 65770

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.6, 41.7, and 41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 38	Tier #	2	2
K/A Statement: Engineered Safety Features Actuation:	Group #	1	1
Operational implications of: Definitions of safety train and	K/A #	013.K5.01	
ESF channel	Importance Rating	2.8	3.2

Proposed Question:

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

1. A steam break occurs on the "A" SG, causing Containment pressure to rapidly rise.
2. Protection Set II CTMT pressure transmitter (3LMS*PT936) fails as-is, and does NOT detect the pressure rise.

Complete the following statement describing required ESF coincidence and ESF Train response:

2 of___(1)____remaining channels must sense the Containment high-pressure condition to generate a Hi-1 Safety Injection signal, and the SIS signal will be sent to SSPS___(2)____.

	(1)	(2)
a)	2	Train "A" only
b)	2	Trains "A" and "B"
c)	3	Train "A" only
d)	3	Trains "A" and "B"

Proposed Answer: B

Explanation (Optional): The coincidence for CTMT Hi-1 SIS is 2 (allowing for one failed channel without a reactor trip) out of 3 signals, since no control functions are tied to CTMT pressure ("C" and "D" wrong). A SIS signal is sent to both trains of RPS for redundancy ("B" correct, and "A" wrong). "C" and "D" are plausible, since SIS signals with control systems (such as Pzr pressure) require 2/4 coincidence. "A" is plausible, since the "B" Train-powered Protection Set II CTMT pressure transmitter failed.

Technical Reference(s): Functional Dwg 8 (Rev K) (Attach if not previously provided)
AOP 3571 (Rev 9-7), Att. R, Page 3

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05493 Describe the operation of the following RPS controls and interlocks… ESF Actuation Signals: 1. Safety Injection… Ctmt Hi-1… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 39	Tier #	2	2
K/A Statement: Engineered Safety Features Actuation	Group #	1	1
Ability to manually operate and/or monitor:	K/A #	013.A4.01	
ESF-initiated equipment which fails to actuate	Importance Rating	4.5	4.8

Proposed Question:

A LOCA has occurred, and initial conditions are as follows:

- The crew is performing actions in E-1, *Loss of Reactor or Secondary Coolant.*
- Both RHR pumps have just been stopped.
- RCS pressure: 600 psia and stable.
- PZR Level: Empty.
- CTMT Pressure: 20 psia
- CTMT Temperature: 175°F
- CTMT Rad levels: 10^4 R/hr

The LOCA increases in size, and the following sequence of events occurs:

Time	Event
0900:	RCS Pressure drops to less than 500 psia.
0908:	CTMT Temperature increases above 180°F.
0916:	CTMT Radiation increases above 10^5 R/hr.
0924:	RCS Pressure drops to less than 300 psia.

When was the crew first required to restart the RHR Pumps?

a) When RCS pressure dropped below 500 psia.

b) When CTMT temperature increased above 180°F.

c) When CTMT radiation increased above 10^5 R/hr.

d) When RCS pressure dropped below 300 psia.

Proposed Answer: B

Explanation (Optional): To provide adequate ECCS flow, RCS pressure should be monitored to ensure that the RHR pumps are manually restarted if pressure decreases to LESS THAN 300 psia ("D" plausible, but wrong, since Adverse Containment conditions came in prior to dropping below 300 psia) (500 psia ADVERSE CONTAINMENT "A" wrong, but plausible, since when pressure initially dropped below 500 psia Adverse Ctmt conditions did not exist). "B" is correct, and "C" wrong, but plausible, since both numbers indicate Adverse Ctmt conditions (>180°F, or 10-5 R/hr), causing the pressure setpoint to increase to 500 psia, but the CTMT temperature condition came in first.

Technical Reference(s):	E-0 (Rev 27-0), Step 15	(Attach if not previously provided)
	E-1 (Rev 26-0), Step 8 Caution	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-07422 Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of E-1.	(As available)

Question Source:	Bank 63963
Question History:	
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.10
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 40	Tier #	2	2
K/A Statement: Containment Cooling:	Group #	1	1
Predict and/or monitor parameters associated with	K/A #	022.A1.03	
operating controls including: Containment humidity	Importance Rating	3.1	3.4

Proposed Question:

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

- Containment temperature is 100°F.
- Containment relative humidity is 40%.

An inadvertent CIA occurs, resulting in the following sequence of events:

1. The crew commences restoring from the inadvertent CIA.
2. The RO reports the CAR Fan Coolers are currently being cooled from the alternate cooling water source.
3. The crew realigns the normal cooling source back to the CAR Fan Coolers.

Correctly complete the following statement on how the cooling source realignment will affect moisture removal from the Containment atmosphere; and where the crew can monitor Containment humidity in the control room.

Realigning the normal cooling source to the CAR Fan Coolers will remove___(1)____moisture from the Containment atmosphere than was being removed while on the alternate cooling water source. Containment humidity can be monitored via a dewpoint indication at___(2)____.

	(1)	(2)
a)	LESS	MB2
b)	LESS	VP1
c)	MORE	MB2
d)	MORE	VP1

Proposed Answer: C

Explanation (Optional): Each CAR fan draws air across the cooling coil assembly and discharges the air to a common duct which distributes it through secondary ducts to different levels of the containment. The crew has just realigned Reactor Plant Chilled Water (CDS) to the CAR fans, replacing RPCCW. Since CDS is colder (about 45°F) than RPCCW (about 85°F), this will increase CTMT cooling, decreasing CTMT temperature. Also, for air at 100°F and 40% relative humidity, the dewpoint is 72°F, so the warmer RPCCW water was above the dew point, and the colder Chilled Water is below the dew point, so the cooling coils will condense water vapor from the CTMT air as it passes over the coils, removing more moisture from the Containment atmosphere ("A" and "B" wrong). Dewpoint meter 3LMS-ME22C indicates Containment humidity on MB2 ("C" correct, "D" wrong). "A" and "B" are plausible, since changing the cooling water temperature affects moisture removal, and two different cooling sources are being considered. "D" is plausible, since VP1 contains Ctmt Ventilation System Controls.

Technical Reference(s):	P&ID 154A (Rev 26)	(Attach if not previously provided)
	OP 3313B (Rev 7-2), Section 1.2	
	OP 3330A (Rev 18-6), Step 4.1.4	
	OP 3330C (Rev 10-2), Section 1.2	
	www.newworldencyclopedia.org/entry/Air_conditioning, Page 40	
	www.dpcalc.org (not attached to this question)	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04253 Describe the purpose of the following Containment Ventilation Sub-systems... Containment Air Recirculation System...	(As available)
Question Source:	Bank 86749	
Question History:	Millstone 3 2011 NRC Exam	
Question Cognitive Level:	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41.4 and 41.9	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Question # 41	Tier #	2	2
K/A Statement: Containment Spray:	Group #	1	1
Bus power supplies to: Containment Spray Pumps	K/A #	026.K2.01	
Proposed Question:	Importance Rating	3.4	3.6

Initial Conditions:

- The plant is at 100% power.
- The "A" Containment Recirculation Pump (3RSS*P1A) is running for a breaker operability check.

The Main Generator Output Breaker spuriously trips open, resulting in a reactor trip.

What is the source of electrical power to the "A" Containment Recirculation Pump?

a) 4160 Volt Bus 34A, which is receiving power from the Normal Station Service Transformer.

b) 4160 Volt Bus 34C, which is receiving power from the Normal Station Service Transformer via the 34A to 34C Cross-Tie Breaker.

c) 4160 Volt Bus 34A, which is receiving power from the Reserve Station Service Transformer via the 34A to 34C Cross-Tie Breaker.

d) 4160 Volt Bus 34C, which is receiving power from the Reserve Station Service Transformer.

Proposed Answer: B

Explanation (Optional): The normal source of power for 4160 Volt Busses 34A and 34C with the plant on line is from the output of the Main Generator, through the "A" Normal Station Service Transformer (NSST), to Bus 34A and 34C. On a loss of the Main Generator, offsite power will back-feed through the GSU Transformers, through the "A" NSST to supply bus 34A and 34A ("C" and "D" wrong). The RSS Pumps are powered from Emergency Bus 34C ("B" correct, "A" wrong). "A" is plausible, since the RSS pumps are supplied by 4160 Volt power and many of the pumps supplied by bus 34A are "A" pumps. "C" and "D" are plausible, since the Generator Output Breaker has tripped open, and the "A" Reserve Station Service Transformer is designed to automatically supply the 4160 Volt Busses on a loss of the "A" NSST.

Technical Reference(s): EE-1A (Rev 25), and 1L (Rev 17) (Attach if not previously provided)
FSAR Section 8.3, pages 8.3-2 and 8.3-4 (Rev 24-4)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03337 Describe the 4kV Distribution System operation under normal, abnormal and emergency conditions... Main Generator Trip... (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 42	Tier #	2	2
K/A Statement: Containment Spray:	Group #	1	1
Monitor automatic operation, including:	K/A #	026.A3.02	
Verification that cooling water is supplied to the containment spray heat exchanger	Importance Rating	3.9	4.2

Proposed Question:

A CDA has occurred, and the crew has entered E-1, *Loss of Reactor or Secondary Coolant.*

The CTMT Recirc (RSS) Pumps have just started, and the crew is transitioning to ES-1.3, *Transfer to Cold Leg Recirculation.*

What is the expected position of the RPCCW heat exchanger Service Water inlet isolation valves (3SWP*MOV50A/B) and the Service Water inlet valves to the Containment Recirculation coolers (3SWP*MOV54A/B/C/D)?

	3SWP*MOV50A/B	3SWP*MOV54A/B/C/D
a)	OPEN	OPEN
b)	OPEN	CLOSED
c)	CLOSED	OPEN
d)	CLOSED	CLOSED

Proposed Answer: C

Explanation (Optional): On a CDA, the RPCCW heat exchanger service water inlet isolation valves receive a CLOSE signal ("A" and "B" wrong), and the service water inlet valves to the containment recirc coolers receive an OPEN signal ("C" correct, "D" wrong). There is a 3 minute time delay prior to 3SWP-MOV54C and D opening on a CDA signal, but since the RSS pumps start on RWST LO-LO level, at least 35 minutes have passed since the CDA actuated. "A" and "B" are plausible, since 3SWP*MOV50A/B remain open on a SIS signal without a CDA, and one of them will be re-opened later in ES-1.3 to restore cooling to the Spent Fuel Pool. "D" is plausible, since there is a time delay on 3SWP*MOV54C and D opening on a CDA, and the crew has not yet aligned for cold leg recirculation.

Technical Reference(s): P&ID 133B (Rev 86) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05718 Describe the operation of the Service Water System under the following normal, abnormal, and emergency conditions... Containment Depressurization Actuation (As available)

Question Source: Bank 69683

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 43	Tier #	2	2
K/A Statement: Main and Reheat Steam:	Group #	1	1
Ability to manually operate and/or monitor:	K/A #	039.A4.04	
Emergency feedwater pump turbines	Importance Rating	3.8	3.9

Proposed Question:

Initial Conditions:

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

The following sequence of events occurs:

1. A control room evacuation is required, and the crew enters EOP 3503, *Shutdown Outside Control Room.*
2. Prior to evacuating the Control Room, the US directs the BOP to confirm the Turbine Driven AFW Pump is running, and if not, to start it at Main Board 5.

What actions, if any, is the BOP operator required to perform with the TDAFW Pump Steam Supply AOVs (3MSS*AOV31A, B, and D) and/or MOVs (3MSS*MOV17A, B, and D) to manually start the TDAFW Pump?

a) Open the Steam Supply AOVs ONLY.

b) Open the Steam Supply MOVs ONLY.

c) Open BOTH the Steam Supply AOVs AND MOVs.

d) NO actions are required. The TDAFW Pump auto-started when the reactor was tripped.

Proposed Answer: A

Explanation (Optional): With the plant initially at 7% power, the crew has already shifted from AFW to Main Feed to supply the SGs. The standby lineup for AFW has the Steam Supply AOVs closed ("B" wrong) and MOVs open ("A" correct, "C" wrong). "B" and "C" are plausible, since the AOVs and MOVs are in series in the steam supply line to the TDAFW Pump. "D" is wrong, since on the reactor trip, the TDAFW Pump does NOT auto-start, since there will be minimal shrink on a trip from low power. "D" is plausible, since the crew does trip the reactor at step 1 of EOP 3503, and normally, the TDAFW Pump does auto start on a trip from high power levels on SG Lo-Lo level due to shrink.

Technical Reference(s): EOP 3503 (Rev 15-3), step 7 (Attach if not previously provided)
P&ID 123A (Rev 55)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04365 Describe the operation of the following Auxiliary Feedwater System components, controls, and interlocks... Turbine Driven Auxiliary Feedwater Pump... (As available)

Question Source: New

Question History:

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 # 44	Tier #	2	2
K/A Statement: Main Feedwater System	Group #	1	1
Predict and/or monitor parameters associated with operating	K/A #	059.A1.07	
controls including: Feed Pump speed normal control	Importance Rating	2.5	2.6

Proposed Question:

The plant is at 48% power, and the following sequence of events occurs:

1. The crew commences placing the second Turbine Driven Main Feed Pump, 3FWS-P2B in service.
2. The BOP operator starts raising 3FWS-P2B speed using Manual Speed Control Switch 3TFC-M1B.

At what approximate speed will 3FWS-P2B initially stop increasing when raising speed with the Manual Speed Control Switch; and how is the BOP operator procedurally required to control "B" TDMFP speed at that point?

a) 3FWS-P2B speed stops increasing at 2200 rpm. The BOP will raise the Manual Speed Control Switch to the high speed stop, and then raise speed using Feed Pump Master Speed Controller 3FWS-SK509B.

b) 3FWS-P2B speed stops increasing at 2200 rpm. The BOP will raise the Manual Speed Control Switch to the high speed stop, and then raise speed using NUS controller 3FWS-SK46B.

c) 3FWS-P2B speed stops increasing at 5000 rpm. At this point, the BOP operator will raise the Manual Speed Control Switch to the high speed stop, and then control speed using Feed Pump Master Speed Controller 3FWS-SK509B.

d) 3FWS-P2B speed stops increasing at 5000 rpm. At this point, the BOP will raise the Manual Speed Control Switch to the high speed stop, and then control speed using NUS controller 3FWS-SK46B.

Proposed Answer: B

Explanation (Optional): The lower set of the NUS speed controller and the manual speed controller controls feed pump speed. The low speed switch for the NUS takes control at about 2200 rpm ("C" and "D" wrong), after which the manual control is taken to the high speed stop to allow full range of control by the NUS controller. "B" is correct, since the NUS controller is used to raise speed to the normal operating speed of 5000 rpm ("C" and "D" plausible) before placing the master speed controller in service ("A" wrong, but plausible).

Technical Reference(s): OP3321 (Rev 17-11), steps 4.4.36 - 4.4.44 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04663 Describe operation of main feed water and steam generator water level control systems under the following normal, abnormal, and emergency conditions... Normal at-power operations while increasing or decreasing power between 25 & 100%... (As available)

Question Source: Bank 69863

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 45	Tier #	2	2
K/A Statement: Auxiliary Feedwater:	Group #	1	1
Effect of a malfunction on: RCS	K/A #	061.K3.01	
Proposed Question:	Importance Rating	4.4	4.6

The reactor trips due to a loss of all Main Feedwater, and the following sequence of events occurs:

1. On the trip, NO Auxiliary Feedwater Pumps can be started, either automatically or manually.
2. The crew enters FR-H.1, *Response to Loss of Secondary Heat Sink.*

In accordance with the WOG Background Document, what fission product barrier challenge exists if there is a delay in restoring core cooling?

a) The RCS will heat up to a point where the PZR PORVs will not be able to adequately depressurize the RCS, and at least partial core uncovery is unavoidable.

b) SG levels will drop to a point where subsequent recovery actions that feed all four SGs at maximum rate could cause multiple U-tube failures.

c) SG levels will drop to a point where subsequent recovery actions could lead to a Pressurized Thermal Shock condition in the Reactor Coolant System.

d) ECCS flow will cause RCS pressure to rise to the point where water is relieved through the PZR Safety Valves, resulting in a Reactor Coolant System breach.

Proposed Answer: A

Explanation (Optional): "A" is correct, since the greatest concern without AFW flow is that the RCS will continue to heat up and pressurize, while RCS Δ Temperature decreases with little heat being removed via the SGs. If allowed to heat up and pressurize to the point where the PORVs start lifting to relieve RCS pressure, opening a PORV will not be able to adequately depressurize the RCS due to high saturation temperature/pressure in the RCS, and insufficient RCS feed flow will occur. The mass loss when PORVs are opened at this point would lead to deep core uncovery and core damage before SI flow could recover the core. "B" is wrong, but plausible, since SG tube failure is a concern when feeding a hot-dry SG, but when this concern exists, operators are directed to feed only one SG at max rate. "C" is wrong, but plausible, since the RCS pressurizes as it heats up (but PTS also requires cooldown stress). Also, several EOPS utilize the strategy of cooling down the RCS at maximum rate (E-3, FR-C.1, and ECA0.0). "D" is wrong, but plausible, since the Pzr Safety Valves are not qualified to pass water, but this is a concern on an inadvertent SIS, and FR-H.1 will open the PORVs, preventing the safety valves from opening. Also, Safety Injection has not actuated.

Technical Reference(s): WOG Bkgd (Rev 2) FR-H.1 Introduction (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04939 Assuming no Operator Action, ANALYZE the events following a Reactor Trip on Loss of Feedwater. (As available)

Question Source: Bank 72415

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 46	Tier #	2	2
K/A Statement: Auxiliary/ Emergency Feedwater:	Group #	1	1
Predict impact and mitigate: Loss of dc power	K/A #	061.A2.03	
Proposed Question:	Importance Rating	3.1	3.4

A plant heatup to normal operating temperature and pressure has just been completed, and initial conditions are as follows:

- The Reactor Trip Breakers are open.
- Both Motor Driven Auxiliary Feedwater Pumps are running.

The following sequence of events occurs:

1. DC Bus 1 loses power.
2. The crew enters AOP 3563, *Loss of DC Bus Power.*

What effect does the loss of DC Bus 1 have on the Turbine Driven Auxiliary Feedwater (TDAFW) Pump; and what actions will AOP 3563 direct the operators to take to regain control of AFW flow?

a) The TDAFW Pump CANNOT be started from MB5. The crew will control AFW flow to the "A" and "D" SG's using the "A" MDAFW Pump path, and flow to the "B" and "C" SG's using the "B" MDAFW Pump path.

b) The TDAFW Pump CANNOT be started from MB5. The crew will control AFW flow to all four SG's using the "B" MDAFW Pump via the discharge cross-connect valves.

c) The TDAFW Pump automatically STARTS. The crew will control AFW flow to the "A" and "D" SG's using the TDAFW Pump path, and flow to the "B" and "C" SG's using the "B" MDAFW Pump path.

d) The TDAFW Pump automatically STARTS. The crew will control AFW flow to all four SG's using the "B" MDAFW Pump via the discharge cross-connect valves.

Proposed Answer: C

Explanation (Optional): On a loss of DC Bus 1, the TDAFW Pump Steam Supply Valves fail OPEN, causing the TDAFW Pump to automatically start ("A" and "B" wrong). Also, the following Flow Control Valves fail open: "A" MDAFW Pump discharge path to the "A" and "D" SGs, and one of two series TDAFW Pump discharge valves to each of the four SGs. Also, the "A" Train AFW cross-connect valve fails closed, and the "A" MDAFW Pump suction valve from the DWST opens. The crew will be directed to isolate the "A" MDAFW paths to SG's "A" and "D", since this is no longer a throttleable path. The crew will control AFW flow to the "A" and "D" SG's using the one functioning TDAFW Pump throttle valve in each path, and flow to the "B" and "C" SG's using the "B" MDAFW Pump path ("C" correct and "D" wrong). "A" and "B" are plausible, since the TDAFW Pump is a backup to the MDAFW Pumps, and DC power to its steam supply valves has been lost. Also, DC power has been lost to several AFW flow control valves. "D" is plausible, since the TDAFW Pump has automatically started, and DC power has been lost to several AFW flow control valves.

Technical Reference(s): AOP 3563 (Rev 10-1), Att. A, Step 2 (Attach if not previously provided)

AOP 3563 (Rev 10-1), Att. A load list, page 5 of 8

P&ID 130B (Rev 47)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05324 Given a failure, partial or complete, of plant air systems, determine effects on the systems and interrelated systems (As available)

Question Source: New

Question History:

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 # 47	Tier #	2	2
K/A Statement: AC Electrical Distribution	Group #	1	1
Effect of a malfunction on: Major system loads	K/A #	062.K3.01	
Proposed Question:	Importance Rating	3.5	3.9

The plant is at 50% power when the following events occur:

- The NSST BACKUP1 TRIP alarm actuates on MB8A.
- The NSST Supply Breaker to bus 34A trips open.

What is the effect of this fault on the major loads supplied by the 4KV Electrical System?

a) A fast-transfer to the RSST occurs. Power is maintained to both Emergency and Non-Emergency loads.

b) A fast-transfer to the RSST occurs. Power is maintained to Emergency loads, but lost to Non-Emergency loads.

c) A slow-transfer to the RSST occurs. Power is restored after a few seconds to both Emergency and Non-Emergency loads.

d) A slow-transfer to the RSST occurs. Power is restored after a few seconds to Emergency loads, but lost to Non-Emergency loads.

Proposed Answer: D

Explanation (Optional): While Primary lockout (Ground or Phase Differential) results in a fast transfer ("A" and "B" plausible), Backup Lockout (Ground or Phase Over-current) attempts to isolate a fault potentially on the non-emergency bus by opening the tie breaker before reenergizing the emergency bus via slow transfer ("D" correct, "A" and "B" wrong). "Fast transfer" closes in the RSST Breaker with the bus tie breaker still closed, keeping all loads energized ("C" plausible); while Slow Transfer trips the crosstie breaker, deenergizing the Non-Emergency Bus and hopefully isolating the fault, prior to the RSST closing in ("C" wrong).

Technical Reference(s): OP 3353.MB8A (Rev 3-2), 2-7 (Attach if not previously provided)
LSK-24-2D (Rev 8)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03333 Describe the operation of 4kV Distribution System controls and interlocks… Fast Transfer… Slow Transfer… (As available)

Question Source: Bank 79948

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 48	Tier #	2	2
K/A Statement:	Group #	1	1
Physical connections and/or cause-effect	K/A #	063.K1.03	
relationship with: Battery charger and battery	Importance Rating	2.9	3.5

Proposed Question:

The Kirk-Key interlock associated with 125VDC Swing Battery Charger 7 is designed to prevent parallel operation of which Chargers/Busses?

a) Swing Charger 7 and Normal Charger 1.

b) Swing Charger 7 and Swing Charger 8.

c) DC Buses 1 and 3

d) DC Buses 5 and 6

Proposed Answer: C

Explanation (Optional): There are three swing battery chargers ("B" plausible). These chargers may be cross connected as follows:
Swing charger 7 (301A-3) can supply either DC buses 1 (301A-1) or 3 (301A-2) ("A" plausible).
Swing charger 8 (301B-3) can supply either DC buses 2 (301B-1) or 4 (301B-2)
Swing charger 9 (301C-2) can supply either DC buses 5 (301C-1) or 6 (301D-1).
The Kirk-key interlocks are provided on the swing chargers to prevent cross-connecting the following 125 VDC buses:
Charger 7: DC bus 1 and 3 ("C" correct, and "A", "B", and "D" wrong).
Charger 8: DC bus 2 and 4
Charger 9: DC bus 5 and 6 ("D" plausible)

Technical Reference(s): EE-1BA (Rev 29) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05015 Describe the 125 VDC distribution system operation under normal, abnormal, and emergency conditions… Placing a battery on equalizing charge… (As available)

Question Source: Bank 64439

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level		RO	SRO
Question # 49	Tier #		2	2
K/A Statement: Emergency Diesel Generator:	Group #		1	1
Design feature/interlock which provides for:	K/A #		064.K4.11	
Automatic load sequencer: safeguards	Importance Rating		3.5	4.0

Proposed Question:

A large-break LOCA occurs, and the crew enters ES-1.3, *Transfer to Cold Leg Recirculation*.

Which valve manipulation will the operators perform that will switch the "A" sequencer to the "Recirculation Mode"?

a) OPEN the SI/CHG Pump Cross-connect Valve (3SIH*MV8807A),
<u>OR</u>
OPEN the Recirculation Spray to RHR Isolation Valve (3RSS*MV8837A).

b) CLOSE the RHR Pump Cross-over Valves (3RHS*MV8716A),
<u>AND</u>
OPEN the RHR to CHG and SI Suction Isolation Valves (3SIL*MV8804A).

c) CLOSE the RHR Pump Cross-over valves (3RHS*MV8716A),
<u>AND</u>
OPEN the SI/CHG Pump Cross-connect Valves (3SIH*MV8807A).

d) OPEN the RHR to CHG and SI Suction Isolation Valves (3SIL*MV8804A),
<u>OR</u>
OPEN the Recirculation Spray to RHR Isolation Valves (3RSS*MV8837A).

Proposed Answer: D

Explanation (Optional): "D" is correct, since opening either MV8804A or MV8807A will signal the "A" sequencer that the plant is in the "Recirculation Mode." "A" and "C" are wrong, since 3SIH*MV8807A operation does not switch the sequencer to the recirc mode. "B" is wrong, since 3RHS*MV8716A operation does not switch the sequencer to the recirc mode. "A", "B", and "C" are plausible, since each of these valve operations are taken when aligning the RCS for Cold Leg Recirculation.

Technical Reference(s): LSK 24-9.4C (Rev 11) (Attach if not previously provided)
ES-1.3 (Rev 15-1), step 3.k and 3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04408 List the seven different modes (including each modes associated input signals) of emergency diesel load sequencer. (As available)

Question Source: Bank 70266

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 50	Tier #	2	2
K/A Statement: Process Radiation Monitoring	Group #	1	1
Design feature/interlock which provides for: Release	K/A #	073.K4.01	
termination when radiation exceeds setpoint	Importance Rating	4.0	4.3

Proposed Question:

Process Radiation Monitor 3HVR-RE12 (Degasifier Area Exhaust Ventilation Radiation Monitor) goes into ALARM.

Which automatic signal does **NOT** occur as a result of this alarm?

a) Gaseous Waste Discharge Valve to the Millstone Stack (3GWS-PV49) receives a close signal.

b) Degasifier effluent to Boron Recovery (3GWS-AOV54 and AOV58) receives a divert signal.

c) Gaseous Drains to the Degasifier (3DGS-AOV57) receives a close signal.

d) Letdown flow to the Volume Control Tank (3CHS*AOV71) receives a divert signal.

Proposed Answer: A

Explanation (Optional): "A" is correct, since 3HVR-RE12 does NOT automatically close 3GWS-PV49. "B", "C", and "D" are wrong, since 3HVR-RE12 causes each of these signals to occur. "B", "C", and "D" are plausible, since the Degasifier normally discharges to the Gaseous Waste System, and 3GWS-PV49 auto-closes on a 3GWS-RE48 Alarm.

Technical Reference(s): AOP 3573 (Rev 18-3), Att. A, page 8 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04727 Describe the operation of the… GWS components controls and interlocks… (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 51	Tier #	2	2
K/A Statement: Process Radiation Monitoring:	Group #	1	1
Knowledge of setpoints, interlocks and automatic	K/A #	073.GEN.2.4.2	
actions associated with EOP entry conditions	Importance Rating	4.5	4.6

Proposed Question:

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

1. A RADIATION ALERT annunciator illuminates on Main Board 2.
2. The crew enters AOP 3573, *Radiation Monitor Alarm Response*.
3. The RO goes to the RMS Console to determine the cause of the alarm.

Which Radiation Monitor has both an automatic action associated with it AND requires the crew to take action using another EOP/AOP in addition to AOP 3573?

a) 3ARC-RE21, Condenser Air Ejector

b) 3CCP-RE31, RPCCW

c) 3LWS-RE70, Liquid Waste Effluent

d) 3SSR-RE08, SG Blowdown Effluent

Proposed Answer: D

Explanation (Optional): "D" is correct, since SSR08 automatically isolates blow down, AND requires the crew to take action per AOP 3576, *Steam Generator Tube Leak*. "A" is wrong, since ARC21 has no automatic action associated with it, but plausible, since ARC21 requires the crew to take action per AOP 3576, *Steam Generator Tube Leak*. "B" is wrong, since CCP31 has no automatic actions associated with it, but plausible, since it requires the crew to take action per AOP 3555, *Reactor Coolant Leak*. "C" is wrong, since LWS70 does not require actions per another AOP, but plausible, since it automatically closes the Liquid Waste Discharge Valve to the Circ Water Tunnel.

Technical Reference(s): AOP 3573 (Rev 18-3), Att A, Pages 1, 11- 12 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05467 Describe the operation of the following Radiation Monitoring System Radiation Monitors Controls and Interlocks… SSR-RE08… (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.10 and 41.11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 52	Tier #	2	2
K/A Statement: Service Water:	Group #	1	1
Effect of a malfunction on ESF loads	K/A #	076.K3.07	
Proposed Question:	Importance Rating	3.7	3.9

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

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

Correctly complete the following statement regarding the status of Charging Pump Cooling (CCE) to the "A" Charging Pump.

Cooling is via feed and bleed cooling, with ___(1)___ being supplied to the ___(2)___ side of the CCE Heat Exchanger.

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

Proposed Answer: C

Explanation (Optional): AOP 3560 directs using Attachments "A" and "B" if a transition to E-0 is made. Attachment "A" has RCP trip criteria, and Attachment "B" aligns feed and bleed cooling to the Charging Pumps, which is required since Charging Pump cooling is normally supplied by Service Water, and has been lost. Attachment "A" aligns Fire Water ("A" and "B" wrong) to the Service Water Side of the CCE Heat Exchanger ("C" correct, "D" wrong). "A", "B", and "D" are plausible, since RPCCW normally supplies makeup to the CCE Heat Exchanger, and supplying feed and bleed cooling to either side of the heat exchanger would cool the Charging Pump.

Technical Reference(s): AOP 3560 (Rev 8-1), note prior to step 2 (Attach if not previously provided)
AOP 3560 (Rev 8-1), Att B, step 2

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03927 Describe the major action categories contained within AOP-3560. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 53	Tier #	2	2
K/A Statement: Service Water:	Group #	1	1
Knowledge of annunciator alarms, indications,	K/A #	076.GEN.2.4.31	
or response procedures	Importance Rating	4.2	4.1

Proposed Question:

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

1. The CHLR A CNDSR SW FLOW HI (MB1C, 4-1A) annunciator is received.
2. The crew enters the associated Annunciator Response Procedure (ARP).
3. A PEO is dispatched, and reports a Service Water (SWP) piping rupture downstream of the "A" Control Building Chiller (3HVK*CHL1A).
4. The BOP Operator stops 3HVK*CHL1A, and a PEO closes the Control Building Air Conditioning Service Water Return Valve (3SWP*V15).

What other valve will the ARP direct the operators to close?

a) The "A" Charging Pump Cooling System Heat Exchanger Return Valve.

b) The "A" Safety Injection Pump Cooling System Heat Exchanger Return Valve.

c) The "A" Emergency Diesel Service Water Return Valve.

d) The "A" MCC/Rod Control Area ACU Return Valve.

Proposed Answer: C

Explanation (Optional): The ARP directs the crew to stop the "A" Emergency Diesel, if running, then close the "A" Emergency Diesel Service Water Return Valve ("C" correct), and go to AOP 3560, *Loss of Service Water* ("A", "B", and "D" wrong). This action is required since the Control Building Chiller and the Emergency Diesel share a common return line prior to discharging to the Circulating Water System. "A", "B", and "D" are plausible, since these are all loads supplied by the "A" Train of Service Water.

Technical Reference(s): OP 3353. MB1C (Rev 6-3), 4-1A, Steps 1- 2 (Attach if not previously provided)
P&ID 133D (Rev 44)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07125 Given a failure, partial or complete, of the Service Water System, determine the effects on the system and interrelated systems. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 54	Tier #	2	2
K/A Statement: Instrument Air:	Group #	1	1
Design feature/interlock which provides for:	K/A #	078.K4.01	
Manual/automatic transfers of control	Importance Rating	2.7	2.9

Proposed Question:

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

1. The plant trips due to a loss of offsite power.
2. The crew enters ES-0.1, *Reactor Trip Response*.
3. The crew resets LOP at MB2 per ES-0.1 direction.
4. The crew manually starts an Instrument Air Compressor at MB1.

Which Instrument Air Compressor did the RO manually start, and would the Air Compressor have started if the crew failed to reset LOP at MB2?

a) The RO started the "A" IAS compressor. The compressor would NOT have started if the LOP signal had not been reset at MB2.

b) The RO started the "A" IAS compressor. The compressor WOULD have started even if the signal had not been reset at MB2.

c) The RO started the "B" IAS compressor. The compressor would NOT have started if the LOP signal had not been reset at MB2.

d) The RO started the "B" IAS compressor. The compressor WOULD have started even if the LOP signal had not been reset at MB2.

Proposed Answer: D

Explanation (Optional): This matches the KA statement about automatic/manual transfer of control since normally, the Instrument Air Compressors operate automatically. Operator action is taken in ES-0.1 on an LOP to have the operators manually start an instrument air compressor from the control room. On an LOP, the EDGs will re-energize the emergency busses. The "A" instrument air (IAS) compressor will not have power available, since it is powered from non-emergency bus 32P ("A" and "B" wrong). The "B" IAS compressor has power ("A" and "B" plausible), but its breaker tripped on the LOP signal. ES-0.1 directs the crew to close the "B" IAS compressor breaker at MB1 to manually restore IAS header pressure. Resetting LOP is not necessary, since the MB2 LOP reset allows manually stopping loads, but the manual start block ("C" plausible) clears automatically 40 seconds after the EDG energizes the bus, and this time has passed well before reaching the step in ES-0.1 ("D" correct, "C" wrong).

Technical Reference(s): ES-0.1 (Rev 25-0), steps 3.d and 3.h (Attach if not previously provided)
OP 3332A-004 (Rev 4-3), page 2
LSK 12-1E (Rev 7), 24.9.4.A (Rev 12)
LSK 24.9.4.B (Rev 12), 24.9.4.P (Rev 10)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05323 Describe operation of plant air systems under the following normal, abnormal, and emergency operating conditions… Loss of offsite power (LOP) (As available)

Question Source: Bank 86756

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 55	Tier #	2	2
K/A Statement: Containment:	Group #	1	1
Ability to manually operate and/or monitor:	K/A #	103.A4.04	
Phase A and phase B resets	Importance Rating	3.5	3.5

Proposed Question:

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

1. The crew enters E-0, *Reactor Trip or Safety Injection*.
2. The crew transitions to ES-1.3, *Transfer to Cold Leg Recirculation*.
3. Per ES-1.3, step 1, the US directs the RO to reset ESF Actuation Signals.

Which procedurally directed sequence is used to reset CIA, CIB, SIS, and CDA to ensure all signals will reset properly?

a) CIA is reset first, followed by a bulleted reset of the remaining signals.

b) CIB is reset first, followed by a bulleted reset of the remaining signals.

c) SIS is reset first, followed by a bulleted reset of the remaining signals.

d) CDA is reset first, followed by a bulleted reset of the remaining signals.

Proposed Answer: C

Explanation (Optional): ES-1.3, step 1 directs the operators to reset SI first (sub-step a), to ensure the CIA reset will function properly. CIB can be reset with CDA still present, so each of the 3 remaining resets (CIA, CIB, and CDA) are bulleted (may be performed in any order) in sub-step b ("A", "B", and "D" wrong). "C" is correct, since alpha-numeric steps are to be performed in sequence. "A", "B", and "D" are plausible, since all four signals are reset in this step.

Technical Reference(s): ES-1.3 (Rev 15-1), step 1 (Attach if not previously provided)
ES-1.3 Step Deviation Doc (Rev 15), step 1

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05493 Describe the operation of the following RPS controls and interlocks… ESF Reset & Block Switches (As available)

Question Source: New

Question History:

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 # 56	Tier #	2	2
K/A Statement: Reactor Coolant:	Group #	2	2
Design feature/interlock which provides for:	K/A #	002.K4.10	
Overpressure protection	Importance Rating	4.2	4.4

Proposed Question:

Both trains of the Cold Overpressure Protection System (COPPS) are ARMED.

A transient occurs, causing RCS pressure to start increasing.

Correctly complete the following statement regarding how Train "B" of COPPS functions to protect the RCS from the overpressure condition.

Train "B" COPPS pressure setpoint is calculated based on auctioneered low___(1)___Leg WR temperature. If actual RCS pressure, as measured by RCS___(2)___, is greater than the COPPS setpoint, PORV 3RCS*PCV456 opens to relieve the over-pressure condition.

	(1)	(2)
a)	Cold	Loop 4 Transmitter 3RCS*PT403
b)	Cold	Loop 1 Transmitter 3RCS*PT405
c)	Hot	Loop 4 Transmitter 3RCS*PT403
d)	Hot	Loop 1 Transmitter 3RCS*PT405

Proposed Answer: A

Explanation (Optional): "A" is correct since Train "B" of COPPS uses Loop 4 WR pressure ("B" and "D" wrong), as measured by 3RCS*PT403 to compare to a pressure setpoint which is calculated based on the current Auctioneered low Cold Leg WR Temperature ("C" wrong), and if necessary sends an open signal to the 'B' PORV (3RCS*PCV456). "A", "B" and "C" are plausible, since Train "A" COPPS uses auctioneered low RCS hot leg temperature, and Loop 1 pressure 3RCS-PT405.

Technical Reference(s): Functional Dwg 19 (Rev D) (Attach if not previously provided)
P&ID 102A (Rev 31)
P&ID 102C (Rev 24)
P&ID 102E (Rev 26)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05334 Describe the function and location of the following Pressurizer Pressure and Level Control System components... Pressurizer Power Operated Relief Valves...Cold Over Pressure Protection System "ARM/BLOCK" Switches (As available)

Question Source: Bank 80573

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41. 3 and 41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 57	Tier #	2	2
K/A Statement: Pressurizer Level Control:	Group #	2	2
Knowledge of the effect of a malfunction of the following	K/A #	011.K6.05	
will have on Pressurizer Level Control:	Importance Rating	3.1	3.7

Function of PZR level gauges as post-accident monitors

Proposed Question:

The reactor has tripped, and the crew has entered the EOP network.

The RO is checking if SI is required, and is currently checking the three Pressurizer level channels on Main Board 4.

What is the meaning of the blue Lamicoid labels under each of the three Pressurizer level indicators on MB4?

a) These indications have instruments at the Auxiliary Shutdown Panel, since they are required to ensure sufficient capability is available from outside the Control Room to shutdown the plant to MODE 3 and cooldown the plant to MODE 5.

b) These indications have backup detectors that can be selected at the Fire Transfer Switch Panel, since they are required to ensure sufficient capability is available to shutdown and cooldown the plant in the event of a fire in any area of the plant.

c) These detectors are designed to continue to function during accident conditions, since this is a key parameter for assessing plant conditions.

d) These detectors are not vulnerable to inaccuracies during accident conditions, and do not require Adverse Containment numbers in the EOPs.

Proposed Answer: C

Explanation (Optional): "C" is correct, and "A" and, "B" wrong, since PAM monitors are designed to continue to function during accident conditions, since they monitor key parameters required for assessing plant conditions during an accident. "A" is plausible, since this is a basis for Remote Shutdown Instrumentation. "B" is plausible, since this is a basis for Fire Related Safe Shutdown indications. "D" is wrong, since adverse containment numbers are applied to Pzr level indication during an accident due to reference leg heating. "D" is plausible, since Pzr level detectors are designed to provide continue to function during an accident.

Technical Reference(s):	Tech Spec Bases for LCO3/4.3.3.6 (Oct 21, 2008)	(Attach if not previously provided)
	SP 3673.6-001 (Rev 9-8)	
	E-1 (Rev 26-0), step 6	
	Process Drawing 11 (Rev J)	

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05340 Describe the major administrative or procedural precautions and limitations placed on the operation of the Pressurizer Pressure and Level Control System, and the basis for each. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7, 41.10, and 43.2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 58	Tier #	2	2
K/A Statement: Rod Position Indication:	Group #	2	2
Predict impact and mitigate: Loss of power to the RPIS	K/A #	014.A2.02	
Proposed Question:	Importance Rating	3.1	3.6

The following sequence of events occurs:

1. The MCC LOSS OF CONTROL POWER annunciator comes in on MB8.
2. The RO reports DRPI indication has lost power on MB4.
3. Upon investigation, MCC 32-2C is found to be de-energized, and cannot be reenergized for at least 12 hours.

What is the status of DRPI "Rod Supervision" indication on the Plant Process Computer; and per OP 3302B, *Digital Rod Position Indication*, how will the crew reenergize DRPI prior to restoring MCC 32-2C?

a) Rod Supervision has also lost indication. Select 32-1C at the Transfer Switch Box in the Aux Building MCC/Rod Control Area.

b) Rod Supervision has also lost indication. Select 32-1M at the Transfer Switch Box in the Aux Building MCC/Rod Control Area.

c) Rod Supervision is still providing DRPI indication. Select 32-1C at the Transfer Switch Box in the Aux Building MCC/Rod Control Area.

d) Rod Supervision is still providing DRPI indication. Select 32-1M at the Transfer Switch Box in the Aux Building MCC/Rod Control Area.

Proposed Answer: B

Explanation (Optional): The power supply for DRPI is selected at 3RDI-TRS1 in MCC Rod Control Area, 45' 6" elevation. The only individual rod position sensors are the DRPI coils, which supply both the DRPI LEDs on MB4 and Rod Supervision. Since DRPI has lost power, DRPI and Rod Supervision no longer display rod height ("C" and "D" wrong). The two available power supplies are 32-2C and 32-1M ("B" correct, and "A" wrong). "C" and "D" are plausible, since Rod Supervision is a separate indication of DRPI. "A" is plausible, since 32-1C is another non-emergency MCC, and power was lost to 32-2C.

Technical Reference(s): OP3302B (Rev 5-5), Section 4.1 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-05298 Describe the function and location of the following Rod Position Indication System components: A. Control Board Display Unit B. Bank Demand Step Counters C. DRPI Power Transfer Switch	(As available)

Question Source: Bank 68722

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 59	Tier #	2	2
K/A Statement: Non-nuclear Instrumentation:	Group #	2	2
Operational implications of: Separation of	K/A #	016.K5.01	
control and protection circuits	Importance Rating	2.7	2.8

Proposed Question:

The Master Pressurizer Pressure controller develops a ground in its control circuitry.

What is the immediate electrical effect, if any, on the Reactor Protection System?

a) No effect, since protection and control circuits use separate detectors, power supplies, and circuitry.

b) No effect, since the control circuit is electrically isolated from the protection circuit.

c) The associated protection system bistable trips.

d) The associated protection system alarms actuate, but the bistable does NOT trip.

Proposed Answer: B

Explanation (Optional): "B" is correct, and "A", "C", and "D" wrong, since control signals going to the controller are electrically isolated from the protection circuits, to prevent electrical faults in control circuits from affecting safety circuits. "A" is plausible, since this relates to circuits used to shutdown the plant when the crew is required to evacuate the control room due to a fire. "C" is plausible, since the circuits use the same input signals. "D" is plausible, since alarm circuits upstream of the trip bistables are not electrically isolated from control circuits.

Technical Reference(s): Process Sheet 12 (Rev N) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05243 Discuss the difference between "process control" and "process protection". (As available)

Question Source: Bank 68324

Question History: Millstone 3 2001 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 60	Tier #	2	2
K/A Statement: Containment Iodine Removal:	Group #	2	2
Bus power supplies to: Fans	K/A #	027.K2.01	
Proposed Question:	Importance Rating	3.1	3.4

A Security Guard calls the Control Room and reports smoke coming from the MCC Cubicle for the "B" Containment Air Filtration Fan (3HVU-FN3B).

Which is the correct MCC to which the Fire Brigade should be dispatched?

a) MCC 32-1M

b) MCC 32-2C

c) MCC 32-3T

d) MCC 32-5H

Proposed Answer: A

Explanation (Optional): "A" is correct, since the power supply to 3HVU-FN3B is MCC 32-1M. "B" is wrong, but plausible, since 32-2C supplies power to the "A" CAF Fan. "C" is wrong, but plausible, since 32-3T supplies loads in the Turbine Building. "D" is wrong, but plausible, since 32-5H supplies "B" Train loads in the Intake Structure.

Technical Reference(s): OP 3313D-001 (Rev 0-0), page 2 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03320 Describe the function and location of the following major 480 volt ac system components… 480 volt MCC's (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 61	Tier #	2	2
K/A Statement: Steam Generator:	Group #	2	2
Design feature/interlock which provides	K/A #	035.K4.02	
for S/G level indication	Importance Rating	3.2	3.5

Proposed Question:

A Main Feedwater transient occurs, resulting in a reactor trip, and current conditions are as follows:

- RCS Tave: 557°F
- Ctmt Temperature: 115°F
- "A" SG Narrow Range Level: 80%
- "B" SG Wide Range Level: 80%
- "C" SG Narrow Range Level: 8%
- "D" SG Wide Range Level: 8%

Based on the various SG level instrument taps, instrument calibrations, and EOP basis information, what is the possible actual status of the Steam Generators?

a) "A" SG may actually be full. "C" SG may actually be empty.

b) "A" SG may actually be full. "D" SG may actually be empty.

c) "B" SG may actually be full. "C" SG may actually be empty.

d) "B" SG may actually be full. "D" SG may actually be empty.

Proposed Answer: D

Explanation (Optional): "D" is correct, since the NR lower tap penetrates the 453 inches (about 38 feet) above the top of the tubesheet, which is also above the top of the U-Tubes, indicating the tubes are covered when on the Narrow Range Scale ("A" and "C" wrong); while the lower WR tap penetrates the downcomer just 21 inches above the top of the tubesheet, and with instrument inaccuracies, 12% Wide Range level may indicate the SG is actually dry ("B" wrong). The upper taps for both the WR and NR detectors are located just above the outlet of the swirl vane separators, 581 inches above the top of the tubesheet. SG Wide Range level is calibrated for COLD conditions (100°F). It will read significantly lower due to less dense water at normal operating temperatures. For example, 50% NR is equivalent to 86% WR level (at 212°F) but with the ruptured SG at the SG safety valve setpoint of 1200 psig, 50% NR is equivalent to 65% WR level. Hash marks on the SG WR level indicators on MB 5 provide operators with a visual queue that SG WR level will indicate about 79% when the top level tap is reached (100% actual level) ("B" wrong). Narrow Range level instruments are calibrated for hot conditions, and indicate accurately at 557°F. "A", "B", and "C" are plausible, since a Wide and Narrow Range Channel share common Upper Level tap.

Technical Reference(s): E-3 (Rev 24-0), step 4 (WR vs. NR Level) (Attach if not previously provided)
FR-H.5 (Rev 8-0), step 4.RNO (Dry SG)
SGS035C (Rev 1-2), page 18
Steam Generator Tech Manual Drawings

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05656 Describe the relationship between Wide and Narrow Range Level Indications in regards to both the top of the Steam Generator Tube Bundle and the Steam Generator Tube Sheet. (As available)

Question Source: New

Question History:

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 # 62	Tier #	2	2
K/A Statement: Steam Dump /Turbine Bypass Control	Group #	2	2
Physical connections and/or cause-effect relationship	K/A #	041.K1.6	
with: Condenser	Importance Rating	2.6	2.9

Proposed Question:

With the plant initially at 100% power, a load reject occurs.

Which valve is specifically designed to aid the Main Condenser in handling the increased energy input during this load reject; and what signal caused this valve to automatically open?

a) The Main Turbine Exhaust Hood Spray Valve (3CNM-TV38), which auto-opens when the Steam Dump Valves arm.

b) The Main Turbine Exhaust Hood Spray Valve (3CNM-TV38), which auto-opens on Condenser high temperature.

c) The De-superheating Spray Valve (3CNM-PV99), which auto-opens when the Steam Dump Valves arm.

d) The De-superheating Spray Valve (3CNM-PV99), which auto-opens on Condenser high temperature.

Proposed Answer: C

Explanation (Optional): The plant is designed to handle a 50% load rejection. 40% is handled by the Main Condenser via the Steam Dumps, and 10% is handled by the Rod Control. In order for the Main Condenser to handle the steam dumped into it from the steam dump system, the De-Superheating Spray valve auto-opens when the steam dumps arm ("C" correct and "D" wrong) to direct condensate pump discharge (cold water) through nozzles in the condenser to cool the steam. This prevents rapid temperature and pressure increases in the condenser during load rejections. The Exhaust Hood Spray Valve opens on exhaust hood high temperature ("D" plausible) to protect against exhaust hood high temperature during low power operation (A" and "B" wrong). "A" and "B" are plausible, since the exhaust hood spray valve is part of the Condensate System, has an AUTO-OPEN feature, and protects against high temperature.

Technical Reference(s): P&ID 126B (Rev 35) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04232 Describe the operation of the following Main Condensate and Makeup Control systems components controls and interlocks... Main turbine exhaust hood spray temperature control valve (3CNM-TV38)... De-superheating spray pressure control valve (3CNM-PV99)... Main condensate system "short recycle" valve (3CNM-FV48)... Low pressure feedwater heater string bypass valve (3CNM-MOV88)... (As available)

Question Source: Bank 69693

Question History:

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 # 63	Tier #	2	2
K/A Statement: Condensate: Physical connections	Group #	2	2
and/or cause-effect relationship with: MFW	K/A #	056.K1.03	
Proposed Question:	Importance Rating	2.6	2.6

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

1. One of the running Main Condensate pumps trips.
2. The BOP operator attempts to start the standby Condensate Pump, but it does NOT start.
3. The BOP operator reports Main Feed Pump suction and discharge pressures are decreasing.

What effect will this have on the Main Feedwater System?

a) The Motor Driven Main Feed pump will auto-start on low feed pump DISCHARGE header pressure. This will restore feed header pressure to normal.

b) A Condensate Demin high differential pressure annunciator will be received at MB6. Operators should promptly open the Condensate Demin Bypass Valve at MB6, restoring feed header pressure to normal.

c) All running Main Feed pumps will trip after a time delay on low feed pump SUCTION pressure. This will result in a LO-LO SG Level reactor trip.

d) A Low Feedwater Pump Suction Pressure annunciator will be received on MB5. Operators should rapidly lower load to 50% using the Load Limit pot at the EHC insert, preventing a reactor trip.

Proposed Answer: C

Explanation (Optional): Each condensate pump is a 50% capacity pump, so two are needed for operation above approximately 50% power ("A" and "B" wrong). "A" is plausible, since the motor driven main feed pump has an AUTO-START feature on low feed pump discharge header pressure. "B" is plausible, since insufficient condensate supply will result in a demin high DP condition, and the Annunciator Response Procedure directs the operators to bypass the demineralizers, increasing suction head to the Main Feedwater pumps. All 3 Main Feed pumps have a trip on low feed pump suction pressure after a time delay. With only 1 condensate pump running, insufficient feed pump suction pressure will lower discharge pressure, and feed pump speed control will speed up the turbine driven main feed pumps, lowering suction pressure even more. The MDMFP may start on low discharge pressure, but this will only lower the suction pressure even more. The feed pumps will trip on low suction pressure, resulting in a reactor trip ("C" correct). "D" is wrong, since rapidly lowering turbine load with the load limit pot will arm steam dumps, which will maintain steam demand even with turbine load lowering. "D" is plausible, since the rapid downpower AOP allows using the load limit pot to rapidly lower load, and 50% is the capacity of one Condensate Pump.

Technical	LSK7-5A (Rev 7), 7-5B (Rev 6), 6-1.1A (Rev 9), and 6-1.1B (Rev 10)	(Attach if not
Reference(s):	OP 3319C (Rev 11-7), Att. 1	previously
	FSAR Table 10.4-3 (Rev 16-1)	provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04235 Given a failure, partial or complete, of the Main Condensate and Makeup Control systems, determine the effects on the systems and on interrelated systems.	(As available)

Question Source:	Bank 73617
Question History:	Millstone 3 2000 NRC Exam
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.5 / 43.5
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 64	Tier #	2	2
K/A Statement: Fire Protection:	Group #	2	2
Knowledge of the effect of a malfunction of fire, smoke,	K/A #	086.K6.04	
and heat detectors on Fire Protection	Importance Rating	2.6	2.9

Proposed Question:

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

1. A Heat Detector at the GSU Transformer shorts out.
2. The Control Room receives a GSU transformer FIRE alarm.

Correctly complete the following statement regarding the effect of the shorted-out Heat Detector.

Deluge water spray ___(1)___ initiate on the transformer, and the remainder of the Fire Detection Zone ___(2)___ considered "functional" per the Technical Requirements Manual (TRM).

	(1)	(2)
a)	DOES	IS NOT
b)	DOES	IS
c)	DOES NOT	IS NOT
d)	DOES NOT	IS

Proposed Answer: A

Explanation (Optional): This is considered RO level since the functionality of the remainder of the string is more of a system function issue than a TRM interpretation, and the information is covered in a procedure caution. A single heat detector in alarm will actuate deluge spray ("C" and "D" wrong). Failure of single heat detector causes the entire zone to become non-functional ("A" correct, "B" wrong). "C" and "D" are plausible, since CO2 actuation requires 2 cross-zoned detectors to alarm to cause an automatic actuation. Also, there are manually actuated fire spray systems in the plant. "B" is plausible, since a failure of many types of detectors in the plant does NOT make the remainder of the associated detectors become inoperable.

Technical Reference(s): OP 3341A (Rev 13-9), Section 1.2 (Attach if not previously provided)
OP 3341D (Rev 15-10), Precaution 3.3
TRM 3.3.3.7 (LBDCR 07-M3-018)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04604 Given a failure, partial or complete, of the Fire Protection, Detection, and Control System, determine the effects on the system and on inter-related systems. (As available)

Question Source: Bank 87713

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 65	Tier #	2	2
K/A Statement: AMSAC:	Group #	2	2
Monitor automatic operation of AMSAC	K/A #	AMSAC.A3.01	
Proposed Question:	Importance Rating	Site Priority	Site Priority

AMSAC has just actuated, and the crew is monitoring the Main Boards to verify it has properly operated.

Which of the following is a **direct** output signal from the AMSAC actuation?

a) Close signal to the Blowdown Flow Control Valves (3BDG-HV20A-D).

b) Start signal to all 3 AFW Pumps.

c) Close signal to the SG Chemical Feed Isolation Valves (3SGF*AOV41A-D).

d) Actuation signal to Feedwater Isolation (FWI).

Proposed Answer: B

Explanation (Optional): AMSAC Actuation trips the Main Turbine, starts all 3 AFW Pumps ("B" correct), and isolates the Blowdown (3BDG*CTV22A-D) and Blowdown Sample (3SSR*CTV19A-D) lines. "A" is wrong since Blowdown Flow Control Valves do not receive a close signal from AMSAC. "C" is wrong since the SG Chemical Feed Valves do not receive a close signal from AMSAC. "A" and "C" are plausible, since AMSAC isolates the Blowdown CTVs and Sample paths. Also, the Blowdown Flow Control Valves auto-close on either a Blowdown Tank high pressure condition or a Main Condenser high pressure condition, and the SG Chemical Feed Valves auto-close on a Feedwater Isolation. "D" is wrong because FWI would result from a Reactor Trip with RCS temperature lowering, and not directly from AMSAC. "D" is plausible, since FWI is normally actuated on a reactor trip signal, and FWI sends a trip signal to the main turbine, and AMSAC will actuate whether or not the reactor trips.

Technical Reference(s): OP 3350 (Rev 6-5), Att. 3 (Attach if not previously provided)
Functional Dwg 15 (Rev L)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04086 Describe operation of AMSAC circuitry including the following… Outputs (As available)

Question Source: Bank 76105

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 66	Tier #	3	3
K/A Statement: Generic:	Group #	1	1
Ability to locate and operate components,	K/A #	GEN.2.1.30	
including local controls	Importance Rating	4.4	4.0

Proposed Question:

With the plant at 100% power, a PEO is being briefed on locally closing a Motor Operated Valve (MOV). The brief includes the following information:

- The valve is a Safety-Related MOV that at this point, is still OPERABLE.
- The MOV has an AUTO-OPEN function.
- The SM has given permission to operate the MOV locally.
- This is a non-emergency situation.
- The PEO will operate the valve in the usual manner by locally depressing the de-clutch lever on the MOV, and then rotating the handwheel in the "closed" direction.

In accordance with OP-AA-100, *Conduct of Operations*, will the crew remove power to the MOV prior to the PEO operating the valve; and after the valve is locally closed, is it still considered OPERABLE?

a) The crew will de-energize the MOV. The valve is declared INOPERABLE until it is again operated electrically.

b) The crew will de-energize the MOV. After local operation, the crew will restore power to the valve, restoring the valve to OPERABLE.

c) The crew will keep the MOV energized. The valve is declared INOPERABLE until it is again operated electrically.

d) The crew will keep the MOV energized. After local operation, the valve is considered OPERABLE.

Proposed Answer: A

Explanation (Optional): This is considered RO level since no interpretation of "below the line" ACTIONs are required in the question, and this is testing a big-picture understanding of the impact of local PEO operations on operability without interpretation of specific plant conditions. The usual way to close an MOV locally is to de-energize the MOV ("C" and "D" wrong), depress the de-clutch lever, and then rotate the handwheel in the "closed" direction. The MOV is declared inoperable until the valve is operated electrically ("A" correct, "B" wrong). "C" and "D" are plausible, since the de-clutch lever will physically allow local operation with power still supplied to the valve motor. "B" is plausible, since the valve still has power.

Technical Reference(s): OP-AA-100 (Rev 25-0), Att. 6, section 2.5 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05081 Discuss the requirements for operating MOVs by hand. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 67	Tier #	3	3
K/A Statement: Generic:	Group #	1	1
Ability to locate control room switches,	K/A #	GEN.2.1.31	
controls, and indications and determine they correctly reflect the desired plant lineup	Importance Rating	4.6	4.3

Proposed Question:

After a mid-cycle reactor trip, a plant startup is in progress per OP 3203, *Plant Startup*, and current conditions are as follows:

- The plant is stable at 12% power.
- The oncoming BOP operator observes the following switch/indicator positions on MB5:

MB5 Switch/Indication	Position
• FW PUMPS P4 TRIP BYPASS Switch:	NORMAL
• 3MSS-N07, Steam Dump "MODE SEL" Switch:	STM PRESS
• Atm Relief Bypass 3MSS*MOV74A Lockout Switch (MB5R):	LOCKOUT
• Feed Isolation Valve 3FWS*MOV35A Position Indication:	GREEN

Which switch position/indication should the BOP operator report as "NOT expected" for current plant conditions, and what is the correct position/indication that should exist?

a) The FW PUMPS P4 TRIP BYPASS Switch, which should be in BYPASS.

b) The Steam Dump "MODE SEL" Switch, which should be in TAVE Mode.

c) The Atm Relief Bypass Valve Lockout Switch, which should be in NORMAL.

d) The Feed Isolation Valve Position Indicator, which should indicate RED.

Proposed Answer: A

Explanation (Optional): With the plant at 12% power, the "FW PUMPS P4 TRIP BYPASS" selector switch should be in BYPASS ("A" correct), since it is not placed in "NORMAL" until power is above 25% power. This is a two position "NORMAL/BYPASS" selector switch, located on MB5, and is used to enable or bypass the Reactor Trip signal which trips the MFW Pumps. The Steam Dump "MODE SEL" Switch should be in the Steam Pressure Mode, since it is not placed in Tave Mode until power is above 15% ("B" is wrong, but plausible). The Atm Relief Bypass Valve Cutout Switch should be in LOCKOUT, since these BTP 9.5-1 Fire Safety cut out switches are normally in bypassed to prevent spurious operation in the event of a "hot short". These switches are switched to "Operate" prior to operating the valves, but the condenser steam dumps are in operation ("C" wrong, but plausible). The Feed Isolation Valve position indicator should indicate GREEN, since these valves are bypassed by the Feed Reg Bypass Valves, and are utilized to isolate the Main Feedwater Regulating Valves while feeding with the bypass valves. They are not opened until the crew shifts to the Main Feed Reg Valves at 25% power ("D" wrong, but plausible).

Technical Reference(s): OP 3204 (Rev 19-0), step 4.1.15 (Attach if not previously provided)

OP 3203 (Rev 20-4), steps 4.3.57.f, 58.b, and 64

GA-26 (Rev 1-0), step 6

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03384 Describe the major action categories contained within OP 3203. (As available)

Question Source: Bank 86763

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 68	Tier #	3	3
K/A Statement: Generic: Knowledge of RO duties in the control room during fuel handling, such as responding to alarms, communications, systems operated in the control room, and supporting instrumentation	Group #	1	1
	K/A #	GEN.2.1.44	
	Importance Rating	3.9	3.8

Proposed Question:

Initial Conditions:

- The plant is in MODE 6
- A new fuel bundle is being lowered into core location E-5.

The following sequence of events occurs:

1. The SHUTDOWN MARGIN MONITOR CHANNEL 1 Annunciator is received on MB4C, 2-2.
2. The RO reports both Source Range channels indicate unexplained flux increase.
3. Fuel movement is suspended.

What additional action is required per OP 3353.MB4C, 2-2?

a) Initiate Control Building Isolation (CBI).

b) Stabilize RCS temperature.

c) Commence boration of the RCS.

d) Withdraw the fuel bundle out of the core.

Proposed Answer: C

Explanation (Optional): "C" is correct, since a loss of shutdown margin is occurring, requiring entry into AOP 3566, and the major action taken in AOP 3566 is immediate boration. "A", "B", and "D" are wrong, since none of these actions are directed in either the ARP or by AOP 3566. "A" is plausible, since this action is directed in EOP 3502, *Fuel Handling Accident*, which is designed for use during fuel handling, and fuel handling operations are in progress, but its entry conditions are not currently met. "B" is plausible, since several Tech Spec actions require the crew to stop all positive reactivity additions, but this is not directed by the ARP, and a cooldown is not likely in progress during refueling ops. "D" is plausible, since this action would undo the action that created the increasing counts condition.

Technical Reference(s): OP 3210B (Rev 10-3), Step 4.2.4 (Attach if not previously provided)
OP 3353.MB4C (Rev 6-12), 2-2, step 6
EOP 3502 (Rev. 8-1), steps 3 and 6
AOP 3572 (Rev. 6-0), step 4

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03960 Identify plant conditions that require entry into AOP 3566, Immediate Boration (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 69	Tier #	3	3
K/A Statement: Generic:	Group #	2	2
Knowledge of the process for making	K/A #	GEN.2.2.6	
changes to procedures	Importance Rating	3.0	3.6

Proposed Question:

The following sequence of events occurs:

1. The RO commences a Surveillance Procedure for the "A" SIH pump.
2. The RO recognizes the flowrate listed in the procedure has not been updated based on an impeller modification made during the previous outage.

Is the surveillance allowed to continue? And what action, if any, is the RO allowed/required to take?

a) Yes. Continue with the surveillance, and if the obtained flowrate ends up being within that specified in the procedure, sign off the surveillance as completed satisfactorily. Then notify the Unit Supervisor and initiate a procedure change.

b) Yes. Notify the Unit Supervisor, and obtain Engineering concurrence. Then with US permission, make an Administrative Change to the procedure and continue with the surveillance.

c) Yes. Notify the Unit Supervisor, and obtain concurrence from a second Senior Reactor Operator. Then with US permission, make an Administrative Change to the procedure and continue with the surveillance.

d) No. The surveillance is required to be stopped, since an Administrative Change is not allowed. Ensure the plant is in a safe condition, notify the Unit Supervisor, and initiate a procedure change.

Proposed Answer: D

Explanation (Optional): An Administrative (pen-and-ink) Change is not allowed, since a change to acceptance criterion is beyond the scope of procedure errors allowing an administrative correction. When there is a procedure discrepancy requiring a procedure change, operators are required to stop the work, ensure the plant is in a safe condition, inform supervision, and initiate a procedure change ("D" correct, "A", "B", and "C" wrong). They can continue the procedure when corrections have been made. "A" is plausible, since it requires flow to be within spec for an approved procedure before completing the surveillance. "B" and "C" are plausible, since these both get supervision involved prior to changing the procedure.

Technical Reference(s): AD-AA-102 (Rev. 9), section 3.6.2 (Attach if not previously provided)
MP-05-DC-SAP01 (Rev 8-6), section 2.3.1-2.3.5

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06771 Outline the process for modifying a document (As available)

Question Source: Bank 86764

Question History: Millstone 3 2011 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 # 70	Tier #	3	3
K/A Statement: Generic:	Group #	2	2
Knowledge of conditions and limits in the	K/A #	GEN.2.2.38	
facility license	Importance Rating	3.6	4.5

Proposed Question:

With the plant initially stable at 100% power, a significant transient occurs, and the board operators report the following parameters:

- Tave: 2.0°F above program
- Pressurizer level: 6% above program level
- Pressurizer pressure: 2270 psia
- All SG NR levels: 46%

In accordance with the precautions of OP 3204, *At Power Operation*, which parameter is currently outside of the band assumed by safety analysis?

a) Tave

b) Pzr Level

c) Pzr Pressure

d) SG NR Level

Proposed Answer: B

Explanation (Optional): The operators are to maintain the following system parameters to ensure plant conditions remain within safety analysis calculations:
Tave within 2.5°F of program ("A" wrong)
Pressurizer level within + or - 5% of program ("B" correct)
Pressurizer pressure between 2225 and 2280 psia ("C" wrong)
Steam generator water level (Narrow Range) between 45 and 55% ("D" wrong)
"A", "C", and "D" are plausible, since each of these parameters contain required bands in OP 3204.

Technical Reference(s): OP 3204 (Rev 19-0), Section 3.1.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05444 Describe the major administrative & procedural precautions and limitations placed on the operation of the Reactor Coolant System… (As available)

Question Source: Modified Bank 83948 (Parent question attached)

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5 and 41.10

Comments:
Original Bank Question (prior to modification) is on the following page.

Examination Outline Cross-reference:	Level	RO	SRO
Question # 71	Tier #	3	3
K/A Statement: Generic: Ability to use radiation	Group #	3	3
monitoring systems, such as fixed monitors, portable	K/A #	GEN.2.3.5	
survey instruments, personnel monitoring equipment, etc.	Importance Rating	2.9	2.9

Proposed Question:

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

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

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

a) 3MSS-RE75, "A" Main Steam Line

b) 3ARC-RE21, Condenser Air Ejector

c) 3MSS-RE80A, "A" Steam Generator N-16

d) 3SSR-RE08, Steam Generator Blowdown

Proposed Answer: B

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

Technical Reference(s): Radiation Monitor Manual pg 35, 38 41, 58 (Rev Feb 16, 2012); P&ID 123A (Rev 55) (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05469 Describe the major administrative or procedural precautions and limitations placed on the operation of the Radiation Monitoring System, including the basis for each. (As available)

Question Source: Bank 75660

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 72	Tier #	3	3
K/A Statement: Generic:	Group #	3	3
Ability to control radiation releases	K/A #	GEN.2.3.11	
Proposed Question:	Importance Rating	3.8	4.3

The plant is initially at 100% power with Control Room Area ACU (3HVC*ACU1A) running, when the following sequence of events occurs:

1. The OUTSIDE ATMOSPH RADIATION HIGH annunciator is received on MB2.
2. The RO verifies 3HVC*RE16A and B are in HI ALARM at the RIC.
3. The crew enters AOP 3573, *Radiation Monitor Alarm Response.*
4. The US directs the BOP operator to verify proper operation of the CBI system at VP1.

Which system alignment is "proper" for this event?

a) Control Room Kitchen Area Exhaust Valve 3HVC*AOV20 is OPEN

b) Control Room Air Inlet Valve 3HVC*AOV25 is CLOSED

c) Control Room Area ACU 3HVC*ACU1A is OFF

d) Control Building Filter Fan 3HVC*FN1B is OFF

Proposed Answer: D

Explanation (Optional): A CBI Signal is generated at the Alarm setpoint for 3HVC*RE16A/RE16B, and causes the Control Building Purge and Kitchen exhaust path valves to close ("A" wrong), the Control Building Filter bypass dampers to close, and the Lead "A" Control Building Filter Fan to automatically start. The "B" Train Fan will start only if the "A" Train Fan fails to start ("D" correct). The CBI signal also sends an open signal to the normally open outside air inlet valves to ensure outside makeup air is brought in to pressurize the envelope via the filter ("B" wrong). The operating ACU maintains Control Room temperature and humidity ("C" wrong) 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. "A" and "B" are plausible, since each of these components are impacted by a CBI. "C" is plausible, since some normal ventilation equipment (Kitchen Exhaust path and Control Room Purge path) realigns on a CBI.

Technical Reference(s): OP 3353.MB2B, 1-7 (Rev 0-0) (Attach if not previously provided)
AOP 3573 (Rev 18-3), Att. A, page 5 of 12
OP 3314F (Rev 23-2), Section 4.13.2
P&ID 151A (Rev 33)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04763 Describe operation of HVC/HVK systems under the following … High radiation detected by HVC*RE16A or B… (As available)

Question Source: New

Question History:

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 # 73	Tier #	3	3
K/A Statement: Generic:	Group #	4	4
Knowledge of EOP layout, symbols, and icons	K/A #	GEN.2.4.19	
Proposed Question:	Importance Rating	3.4	4.1

Safety Injection has actuated, and the crew is progressing through the EOP network.

What are the general rules for step completion while in the EOP network?

a) The crew is allowed to proceed to the next step before completing the current step. If the steps must be completed before proceeding, a CAUTION prior to step 1 of the EOP will explicitly state the requirements for that EOP.

b) The crew is allowed to proceed to the next step before completing the current step. If a step must be completed before proceeding, either the logic of the step, a WHEN/THEN statement, or a CAUTION will prevent proceeding.

c) The crew is required to complete the current step before proceeding to the next step. If a step need not be fully completed before proceeding to the next step, a NOTE prior to step 1 of the EOP will explicitly state the allowance to proceed for that EOP.

d) The crew is required to complete the current step before proceeding to the next step. If a step need not be fully completed before proceeding to the next step, either the logic of the step or a NOTE will state the allowance.

Proposed Answer: B

Explanation (Optional): Unless otherwise specified, a step need not be fully completed before proceeding to the next step ("C" and "D" wrong). Once a step is begun, the SM/US may determine it desirable and acceptable to continue the procedure actions even though the current task is not yet complete; however, completing the task in a timely manner is still required. If a particular step or portion of the procedure must be completed prior to proceeding, one of the following methods are used to alert the Operator of this situation: A "CAUTION" is provided explicitly stating this requirement, an RNO is provided with a WHEN, THEN logic statement which prohibits proceeding to the next step until the condition is satisfied, or the logic of the step prevents the Operator from proceeding until a specific condition is satisfied ("A" wrong and "B" correct). "A", "C", and "D" are plausible, since these rules are close to GOP requirements.

Technical Reference(s): OP 3272 (Rev 8-12), Section 1.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04449 Describe when actions of a step need not be fully completed prior to proceeding to the next step within the same procedure or transitioning to another procedure. (As available)

Question Source: Bank 70163

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level		RO	SRO
Question # 74		Tier #	3	3
K/A Statement: Generic: Knowledge of events related		Group #	4	4
to system operation/status that must be reported		K/A #	GEN.2.4.30	
to internal organizations or external agencies, such as		Importance Rating	2.7	4.1
the State, NRC, or transmission system operator				

Proposed Question:

The crew is preparing to conduct a planned load change.

In accordance with OP-AP-300, *Reactivity Management*, what is the minimum load change above which the crew is required to notify Reactor Engineering?

a) 1%

b) 5%

c) 10%

d) 15%

Proposed Answer: B

Explanation (Optional): "B" is correct, and "A", "C", and "D" wrong, since the crew is responsible to notify Reactor Engineering of all planned reactivity changes > 5%. "A", "C", and "D" are plausible, since each of these are power changes that could be planned for plant repairs, etc.

Technical Reference(s): OP-AP-300 (Rev 16-0), Section 3.3.3 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06343 Understand Reactivity Management principles as outlined in OP-AP-300, Reactivity Management. (As available)

Question Source: Bank 67787

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 75	Tier #	3	3
K/A Statement: Generic:	Group #	4	4
Ability to diagnose and recognize trends	K/A #	GEN.2.4.47	
utilizing appropriate reference material	Importance Rating	4.2	4.2

Proposed Question:

With the plant initially at 50% power with rod control in manual, the following sequence of events occurs:

1. A plant transient occurs.
2. The RO reports primary plant parameters to the crew.
3. The STA compares the parameters against the expected parameter values listed in SP 31002, *Plant Calorimetric*, Attachment 9.

The crew determines major parameters have stabilized relative to their initial values as follows:

- Reactor Power (based on calorimetric): Higher than prior to the transient.
- Tave: Returned to its original value.
- MWe: Lower than prior to the transient.

What event could have caused this transient?

a) A dilution is occurring.

b) A rod has dropped.

c) Extraction steam has been lost to a feed heater.

d) A turbine control valve has failed fully open.

Proposed Answer: C

Explanation (Optional): "C" is correct, since MWe changing opposite of reactor power with Tave constant is indicative of a change in plant efficiency. Due to loss of efficiency, T_{cold} will go down and T_{hot} will go up, but Tave will be fairly stable. "A" is wrong, since with a dilution, MWe would not be decreasing. "A" is plausible, since reactor power is increasing. "B" is wrong, since with a dropped rod, Reactor Power would be decreasing. "B" is plausible, since MWe had decreased. "D" is wrong, since with increased steam demand, Tave would be decreasing. "D" is plausible, since reactor power would go up.

Technical Reference(s):	NRC IN 96.41 Comanche Peak 1996	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-03349 For given plant conditions, qualitatively state the effect of any RCS, secondary plant or reactivity induced transient… on the following parameters (RCP trip, turbine trip, dropped rod, etc.): reactor power, rod position, RCS loop average temperatures…	(As available)

Question Source:	Bank 65340
Question History:	
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.1, 41.5, and 41.14
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 76	Tier #		1
K/A Statement: RCP Malfunctions:	Group #		1
Ability to explain and apply system limits	K/A #	APE.015/17.GEN.2.1.32	
and precautions	Importance Rating		4.0

Proposed Question:

The plant is at 20% power, preparing to place the main turbine in service, when the following sequence of events occurs:

1. The "A" RCP # 1 seal leakoff flow increases from 2.3 gpm to 7.2 gpm over 3 minutes.
2. The STA reports both Seal Water inlet temperatures are reading 113°F, and slowly increasing.

What action is required to be directed by the US?

a) Trip the Reactor, stop the "A" RCP, and go to E-0, *Reactor Trip or Safety Injection*.

b) Enter AOP 3554, *RCP Trip or Stopping a RCP at Power*, remove the RCP from service and isolate the number 1 seal within 5 minutes.

c) Enter OP 3206, *Plant Shutdown*, and commence an orderly plant shutdown and remove the "A" RCP from service within 8 hours.

d) Per the RCP Seal ARP, notify the Duty Officer and request Engineering Department evaluate continued pump operation.

Proposed Answer: B

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and select a procedure based on decision points in the ARP based on the specific event. The RCP seal leakoff limit before a trip is required is 8 gpm, due to concern of hot RCS water flowing up the RCP shaft into the seal package. A backup check is seal inlet temperature increasing, since some flow is also exiting the number two seal. Table 1 directs actions of step 6 to be taken, which goes to AOP 3554 to perform an immediate RCP shutdown ("B" correct). Power is below P-8, therefore trip is not required ("A" wrong). , "A" is plausible, since this action would be correct if power were above P-8. "C" is plausible; since an orderly plant shutdown would be performed if seal inlet temperature was stable. "D" is wrong, but plausible since these actions would be correct if seal leakoff flow was above the alarm setpoint but ≤ 6 gpm.

Technical Reference(s): OP 3353.MB3B (Rev 6-21), 2-10, steps 4, 5, and 6 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07533 Given a set of plant conditions, determine the required actions to be taken per AOP 3554. (As available)

Question Source: Bank 64317

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 77	Tier #		1
K/A Statement: Loss of RHR System:	Group #		1
Ability to recognize system parameters that are entry-level conditions for Technical Specifications	K/A #	APE.025.GEN.2.2.42	
	Importance Rating		4.6

Proposed Question:

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

1. During testing of the "B" RHR Pump, vibration is noted to be above normal levels.
2. The System Engineer determines vibration is within limits, and the pump is OPERABLE.
3. The System Engineer recommends tagging the pump out for bearing inspection/repair at the next available opportunity.
4. Maintenance estimates the work will take 80 hours.
5. Management is deciding whether to shut down the plant to perform repairs, or to wait for the upcoming scheduled shutdown for the refueling outage to perform the work.

If a plant shutdown is commenced, what is the first plant MODE in which the pump can be removed from service WITHOUT having to enter an LCO ACTION for the RHR Pump?

a) MODE 3.

b) MODE 4.

c) MODE 5.

d) MODE 6.

Proposed Answer: B

Explanation (Optional): Question is considered SRO level since it requires the applicant to apply four applicable Tech Spec LCOs in lower operational modes. The four LCO's that apply are: LCO 3.4.1.2 (RCS Loops) which requires only RCS loops, until two trains of RHR are required to be OPERABLE in MODE 5 ("D" wrong); LCO 3.5.2 (ECCS), which requires two trains of ECCS in MODES 1, 2, and 3; LCO 3.5.3, which requires only one train in MODE 4 (not provided as a reference), meaning RHR can be removed from service once MODE 4 is reached ("C" correct, "A" and "B" wrong); and LCO 3.9.8.1, which requires one RHR loop in MODE 6, but question asks for first MODE where pump can be removed from service ("D" wrong, but plausible). "A" and "B" are plausible, since for each of these MODES, at least one of the LCOs does not require an RHR pump.

Technical Reference(s): Tech Spec LCO 3.4.1.2 (Amend 230) (Attach if not previously provided)
Tech Spec LCO 3.5.2 (Amend 103)
Tech Spec LCO 3.5.3 (Amend 157)
Tech Spec LCO 3.9.8.1 (Amend 230)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05460 Given a plant condition or equipment malfunction, use provided reference material to... Evaluate Technical Specification applicability... (As available)

Question Source: Bank #64247

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 78	Tier #		1
K/A Statement: Pressurizer Pressure Control Malfunction:	Group #		1
Determine/interpret RCP injection flow.	K/A #	APE.027.AA2.14	
Proposed Question:	Importance Rating		2.9

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

1. The Pressurizer Master Pressure Controller malfunctions, and actual RCS pressure starts to decrease.
2. The RO takes manual control of the Master Pressure Controller, and stabilizes RCS pressure.
3. The RO reports current plant parameters are stable at the following values:
   - RCS Pressure: 2240 psia
   - "A" RCP Seal Injection Flow: 10.1 gpm
   - "B" RCP Seal Injection Flow: 9.9 gpm
   - "C" RCP Seal Injection Flow: 10.1 gpm
   - "D" RCP Seal Injection Flow: 10.1 gpm

What ACTION, if any, is required to be taken due to RCS CONTROLLED LEAKAGE, and what is the basis for the controlled leakage portion of the RCS Leakage LCO?

a) Reduce leakage to within limits within 4 hours or be in HOT STANDBY within the next 6 hours. The basis is to detect RCP seal degradation and take action prior to further degradation which could lead to a LOCA.

b) Reduce leakage to within limits within 4 hours or be in HOT STANDBY within the next 6 hours. The basis is to ensure SI flow is not less than that assumed in the safety analysis in the event of a LOCA.

c) No ACTION is required to be taken. The basis is to detect RCP seal degradation and take action prior to further degradation which could lead to a small break LOCA.

d) No ACTION is required to be taken. The basis is to ensure SI flow is not less than that assumed in the safety analysis in the event of a LOCA.

Proposed Answer: B

Explanation (Optional): Question is considered SRO level since it requires the applicant to apply surveillance requirements and provide Tech Spec Basis information for these specific plant conditions. The basis of the limit is to ensure that during a LOCA, the SI flow will not be less than assumed in the safety analyses ("A" and "C" wrong). Controlled leakage is determined under a set of reference conditions, specifically one Charging Pump in operation and RCS pressure at 2250 psia +/- 20 psia. The cause of the increased injection flowrate is a drop in pressure caused by the Pzr pressure control malfunction, but the pressure dropped by less than 20 psia, so the LCO Action still applies. The combined flow of the four RCPs is 40.1 gpm, which exceeds the LCO limit of 40 gpm, and the ACTION for exceeding controlled leakage is to reduce leakage to within limits within 4 hours or be in HOT STANDBY within the next 6 hours ("B" correct, "D" wrong). "A" and "C" are plausible, since this is related to the basis for identified leakage, and RCP seals are the weak link on a loss of AC power. "D" is plausible, since RCS pressure is low, and flow is close to the controlled leakage limit.

Technical Reference(s):	Tech Spec Bases for 4.4.6.2.e (LBDCR No. 07-MP3-032) Tech Spec Surveillance Req 4.4.6.2.1.c (Amend 238) Surveillance 3601F.3-001 (Rev 4-2), page 2 of 2	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05343 Given a plant condition or equipment malfunction… evaluate Technical Specification applicability and determine required actions.. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 79	Tier #		1
K/A Statement: Loss of Vital AC Inst. Bus	Group #		1
Determine/interpret Instrument Bus alarms for	K/A #	APE.057.AA2.06	
the inverter and alternate source	Importance Rating		3.7

Proposed Question:

The plant is at 100% power, an earthquake occurs, and the following annunciators are received:

INVERTER 3 TROUBLE (MB8A, 1-7)

A PEO is dispatched to investigate, and the PEO reports the following:

- The "Out of Sync Light" is ON
- The "Bypass Source Supplying Load" Light is ON
- The "Inverter Supplying Load" Light is OFF
- The Battery Charger 3 DC Output Breaker has tripped OPEN.

Using the attached copies of Technical Specification LCOs 3.8.2.1 and 3.8.3.1, how long does the crew have from the initiation of the event to restore the normal electrical lineup before having to apply the requirement to "be in HOT STANDBY within the next 6 hours"?

a) 1 hour.

b) 2 hours.

c) 8 hours.

d) 24 hours.

Proposed Answer: D

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions, and apply required Tech Spec actions with specific input from the initial conditions affecting the required action time. An assessment of the alarms shows power is being supplied to the VIAC from the alternate AC source as indicated by the Bypass Source Supplying Load Light ON. Also, since a Charger DC output breaker has tripped open, Battery 3 is no longer being supplied by a Charger. Per LCO 3.8.2.1, the crew has 24 hours to provide a charger for Battery Bank 3. Per 3.8.3.1, there is a 2-hour requirement to energize a deenergized VIAC. Since the VIAC is energized from the alternate source this action is met ("B" wrong, but plausible), so the crew has 24 hours to energize the VIAC from its inverter per ACTION 3.8.3.1.b (2) ("D" correct, "C" wrong). If not, the crew has 2 hours to energize the VIAC per ACTION 3.8.3.1.b (1). "A" is wrong, but plausible, since this is the time allowed by 3.0.3, and multiple failures occurred. "B" is plausible, since this is the time allowed per ACTION 3.8.3.1.c if the DC bus is not energized, and the time allowed to restore a battery bank or charger per 3.8.2.1 if DC Bus 1 or 2 was affected. "C" is plausible since this is the time allowed if an emergency bus (VIAC 3 is a "vital" bus) is not OPERABLE per 3.8.3.1.a.

Technical Reference(s):	Tech Spec LCO 3.8.2.1 (Amend. 64)	(Attach if not previously provided)
	Tech Spec LCO 3.8.3.1 (Amend. 220)	
	OP 3345B (Rev 11-2), section 4.18	
	EE-1BA (Rev 29)	

Proposed references to be provided to applicants during examination: **LCOs 3.8.2.1 and 3.8.3.1**

Learning Objective: MC-03951 Given a plant condition requiring the use of AOP-3563, identify applicable technical specification action requirements (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 80	Tier #		1
K/A Statement: Loss of Nuclear Svc Water:	Group #		1
Ability to determine operability and/or	K/A #	APE.062.GEN.2.2.37	
availability of safety related equipment	Importance Rating		4.6

Proposed Question:

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

1. The Outside Rounds PEO reports a significant motor oil leak exists on the "A" (lead) Service Water Pump.
2. The crew starts the "C" SWP Pump.
3. The crew stops the "A" SWP Pump and places it in Pull-To-Lock.

Which choice below correctly describes the effect of the loss of the Service Water Pump on "A" Emergency Diesel Generator OPERABILITY?

a) The EDG IS still OPERABLE, since there is still an OPERABLE Service Water Pump available on the "A" Train of Service Water.

b) The EDG IS still OPERABLE, since loss of a Service Water Pump is covered by the SWP LCO, which has a more restrictive ACTION time than the EDG LCO.

c) The EDG is NOT OPERABLE because both Service Water pumps are required on the affected Train to maintain OPERABILITY of the EDG.

d) The EDG is NOT OPERABLE because the lead Service Water pump is required on the affected Train to maintain OPERABILITY of the EDG.

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to apply Tech Spec Basis knowledge to analyze operability. The EDG requires an operable SWP "loop". The EDG is operable since only one redundant SWP pump in its train is required to consider the loop operable per Tech Specs ("A" is correct, and "C" and "D" wrong). "C" and "D" are plausible, since the lead Service Water Pump was lost, and it receives the first start signal on an LOP. Also, there are TRM actions with a loss of one SWP Pump, but this is not related to SWP or EDG OPERABILITY. "B" is wrong, since if SWP is lost to a train, its EDG is INOPERABLE, since its LCO time is more restrictive than the SWP LCO. "B" is plausible, since unless the second ACTION time is more restrictive, Tech Spec LCOs are generally not cascaded.

Technical Reference(s): Tech Spec LCO 3.7.4 (Amend. 206) (Attach if not previously provided)
Tech Spec Basis for LCO 3.7.4 (LBDCR 3-22-02)
Tech Spec LCO 3.8.1.1 (Amend. 229)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05720 (SRO, STA) Given a plant condition or equipment malfunction (related to Service Water), use provided reference material to.... Evaluate Technical Specification applicability and determine required actions (As available)

Question Source: Bank #69353

Question History:

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
K/A Statement: Generator Voltage and Electric Grid Disturbances: Determine/interpret VARS outside the capability curve	Group #		1
	K/A #	APE.077.AA2.04	
	Importance Rating		4.0

Proposed Question:

Initial Conditions:

- A load increase is in progress per OP 3204, *At Power Operation*.
- Reactor power is 40% and increasing normally.

An electrical disturbance occurs, and the following sequence of events occurs:

1. The GENERATOR OVER EXCITATION (MB7C, 5-5) Annunciator is received.
2. The BOP operator reports VARS are reading 925 VARS Out.
3. Reactor Power is steady at 40%.
4. The GEN CORE MONITOR LEVEL HI (MB7C, 4-5) Annunciator is received.
5. A PEO is dispatched to the Core Monitor to press and hold the FILTER pushbutton for 15 seconds, and report the results to the Control Room.
6. The PEO reports the following at the Core Monitor:
    - The trace had rapidly declined from normal to the alarm condition.
    - The trace returned to normal when "Filter" was depressed.
    - The trace returned to the alarm condition when the Filter pushbutton was released.

Assuming the reactor does not trip during the transient, what actions are required to be taken by the US?

a) Per OP 3353. MB7C, 5-5, direct the BOP to lower Main Generator voltage; and maintain the Main Generator Voltage Regulator in "Manual" until the problem is corrected.

b) Per OP 3353. MB7C, 5-5, direct the BOP to lower Generator load to reduce Stator current; and maintain the Main Generator on "Load Limit" until the problem is corrected.

c) Per OP 3353. MB7C, 4-5, direct the BOP to trip the Main Turbine and enter AOP 3550, *Turbine/Generator Trip*. After completing AOP 3550, transition to OP 3206, *Plant Shutdown*.

d) Per OP 3353. MB7C, 4-5 direct the BOP to trip the Main Turbine and enter AOP 3550, *Turbine/Generator Trip*. After completing AOP 3550, transition to OP 3207, *Reactor Shutdown*.

Proposed Answer: D

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions, make decisions based on specific procedure decision points, and select the appropriate procedure based on the event in progress. Operating outside the capability curve leads to over-heating, which caused the core monitor alarm. Since the Core Monitor trace decreased rapidly and responded to the filter being placed in service, an actual overheating event is occurring. "A" and "B" are wrong, since the ARP requires a turbine trip and entry into AOP 3550 if less than P-9. "A" and "B" are plausible, since if the trace had decreased gradually, or had not responded to the filter being placed in service, the US would be required to notify I&C of a core monitor malfunction; and the over excitation ARP directs the crew to lower MVARs by reducing main generator voltage. Also, reducing real load reduces Generator current, and "Load Limit" control allows lowering load without relying on Load Set operation. "D" is correct, and "C" wrong, but plausible, since AOP 3550 takes the actions performed by OP 3206 for shutting down the turbine/generator, and transitions the crew directly to OP 3207.

Technical Reference(s): OP 3353.MB7C (Rev. 4-4), 5-5, step 1 (Attach if not previously provided)

OP 3353.MB7C (Rev. 4-4), 4-5, Note 3 and step 4

AOP 3550 (Rev 8-0), step 17

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04685 Describe operation of Main Generator exciter and regulator system under… Generator overload operations… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 82	Tier #		1
K/A Statement: Loss of Source Range NI:	Group #		2
Determine/interpret maximum allowable	K/A #	APE.032.AA2.07	
channel disagreement	Importance Rating		3.4

Proposed Question:

The crew is preparing to start up the reactor per OP 3202, *Reactor Startup*; and the following sequence of events occurs:

1. The crew closes the Reactor Trip Breakers and commences withdrawing Control Rods.
2. While looking for proper overlap between the Source and Intermediate NIS channels, the RO reports Source Range Channel N31 has drifted low.
3. The RO reports NIS Channels indicate as follows:
   - SR Channel N31: 10,000 cps
   - SR Channel N32: 60,000 cps
   - IR Channel N35: $1.2\text{x}10^{-10}$ Amps
   - IR Channel N36: $1.2\text{x}10^{-10}$ Amps
4. The SM determines this exceeds the maximum allowable channel disagreement, and the crew enters AOP 3571, *Instrument Failure Response*.

Correctly complete the following statement regarding whether the startup is allowed to continue per AOP 3571, and the FSAR's crediting the Source Range detectors during the startup.

The startup is__(1)___, and the FSAR___(2)____credit the Source Range NIS as primary protection against a positive reactivity addition event up to this point in the reactor startup.

	(1)	(2)
a)	required to be stopped	DOES NOT
b)	required to be stopped	DOES
c)	allowed to continue	DOES NOT
d)	allowed to continue	DOES

Proposed Answer: C

Explanation (Optional): Question tests the second half of the K/A statement. Question is considered SRO level since it requires assessment of plant conditions related to Technical Specification requirements and determine action required per an AOP, and detailed knowledge of FSAR assumptions. Based on conditions given in the stem, it can be determined that the plant is in MODE 2, and also above P-6. Tech Spec 3.3-1 only requires the Source Ranges in MODE 3 and MODE 2 after the reactor trip breakers are closed with power below the P-6 setpoint of 10^{-10} Amps in the intermediate range. Since power is above P-6, the AOP allows the crew to continue the startup, even with the SRs were still energized, as long as the failed SR is placed in bypass and both Source Range block switches are actuated ("A" and "B" wrong). "A" and "B" are plausible; since the startup would be required to be stopped per AOP 3571 if power were below P-6. Also, the FSAR credits the Power Ranges during the Startup. The FSAR does not credit the Source Ranges, since they are not seismically qualified ("C" correct, "D" wrong). Operators are required to verify RCS Tave is greater than 551°F (minimum temperature for criticality) with the Power Range Startup Surveillance current prior to closing the trip breakers on a reactor startup. "D" is plausible, since LCO 3.3-1 requires Source Ranges to be operable after closing the trip breakers, and does not require Power Ranges in MODE 3.

Technical Reference(s):	AOP 3571 (Rev 9-7), Att. F, step 1	(Attach if not previously provided)
	Tech Spec Table 2.2-1 (Amend. 217), page 2-7	
	FSAR 15.4.1.2.4 (Rev. 26.1)	
	FSAR 9.4.7.3.3 (Rev. 26.1)	
	Tech Spec 3.3.1 (Amend. 229), page 3/4 3-2	
	Tech Spec 3.3.1 (Amend. 217), pages 3/4 3-5	
	Tech Spec Bases Page B 2-5 (LBDCR No. 07-MP3-017)	
	OP 3302A (Rev. 15.7), Prerequisite 2.1.6	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04900 List automatic Control & Protection System actions anticipated as a result of reactivity & power distribution anomalies	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.1, 43.2, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 83	Tier #		1
K/A Statement: Accidental Gaseous Radwaste Rel:	Group #		2
Knowledge of abnormal conditions procedures	K/A #	APE.060.GEN.2.4.11	
Proposed Question:	Importance Rating		4.2

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

1. A RADIATION ALERT annunciator comes in on MB2.
2. The RO reports that the Turbine Building Stack Rad Monitor 3HVR10B has trended up to the ALERT setpoint.

What procedure is the US required to use to mitigate the event, and what action will a specific step in that procedure direct?

a) The US will use AOP 3573, *Radiation Monitor Alarm Response*, which will direct the crew to manually start one train of SLCRS.

b) The US will use AOP 3573, *Radiation Monitor Alarm Response*, which will direct the crew to manually start one train of Auxiliary Building Filtration for general area lower levels.

c) The US will use OP 3353.MB2B, 2-8, *Radiation Alert*, which will direct the crew to manually start one train of SLCRS.

d) The US will use OP 3353. MB2B, 2-8, *Radiation Alert*, which will direct the crew to manually start one train of Auxiliary Building Filtration for general area lower levels.

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to interpret radiation readings as they pertain to selection of the appropriate Abnormal Operating Procedure. "C" and "D" are wrong, since OP3353.MB2B, 2-8 will direct the crew to AOP 3573 prior to taking actions with ventilation. "C" and "D" are plausible, since OP 3353.MB2B, 2-8 is applicable in this event, and will direct some actions. "A" is correct, and "B" is wrong, since the crew is directed to start a SLCRS fan, not the Aux Building Filters. There are other building ventilation systems that discharge out the Turbine Building Stack besides the Aux Building. "B" is plausible, since Aux Building Filters are started for numerous Rad Monitor alarms, and the Aux Building does discharge through the Turbine Building Stack.

Technical Reference(s): AOP 3573 (Rev. 18-3), Att. A, page 7 of 12 (Attach if not previously provided)
OP 3353.MB2B 2-8, (Rev. 0-1)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07567 Given a set of plant conditions, determine the required actions to be taken per AOP 3573. (As available)

Question Source: Bank #80930

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.4 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 84	Tier #		1
K/A Statement: Steam Generator Over-pressure:	Group #		2
Determine/interpret adherence to appropriate	K/A #	W/E13.EA2.2	
procedures	Importance Rating		3.4

Proposed Question:

The reactor trips, resulting in the following initial sequence of events:

1. The crew enters ES-0.1, *Reactor Trip Response*.
2. A yellow path is received due to an overpressure condition in "B" SG.
3. The crew enters FR-H.2, *Response to Steam Generator Overpressure*.
4. The crew is unsuccessful in attempts to release steam from the "B" SG.

The RO reports Pressurizer level has decreased to 8%.

In accordance with the EOP rules of usage, what action is the US required to direct?

a) While in FR-H.2, apply the Foldout Page of ES-0.1, actuate Safety Injection, and then go to E-0, *Reactor Trip or Safety Injection*.

b) Enter ES-0.0, *Rediagnosis*, and then actuate Safety Injection and go to E-1, *Loss of Reactor or Secondary Coolant*.

c) Return to ES-0.1, and then actuate Safety Injection and go to E-1, *Loss of Reactor or Secondary Coolant*.

d) While in FR-H.2, dump steam from the "A", "C", and "D" SGs to decrease "B" SG pressure by cooling it down.

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and select a procedure while integrating EOP rules of usage concerning status tree implementation and use of ES-0.0, *Rediagnosis*. "A" is correct, and "C" and "D" wrong, since while performing a yellow path procedure, an ES procedure that was in effect has priority over the yellow path procedure, and its foldout page items are still applicable. So, based on the ES-0.1 foldout page, with Pressurizer level less than 9%, the crew is required to actuate SI and go to E-0. "C" is plausible, since foldout page items from an ES procedure are not in effect if a orange or red path procedure is in use, and a yellow path procedure can be exited at any time based on changing plant conditions or operator judgment. "B" is wrong, since Rediagnosis is not allowed to be implemented unless SI has already been actuated. "B" is plausible, since ES-0.0 is implemented based on operator judgment after exiting E-0 in events where the current procedure is not taking the desired path to mitigate the event. And, based on rules of usage, the ES-0.1 foldout page would not be in effect if the status tree were orange or red. Also, if the crew did enter ES-0.0 given these conditions, it would direct them to E-1. "D" is plausible, since for orange or red paths, fold out page items from the previous ES procedure are not in effect, and the crew is required to complete the FR procedure prior to returning to the ES procedure. And per FR-H.2, if unsuccessful at dumping steam from the affected SG, the crew is directed to dump steam from the unaffected SGs to reduce RCS hot leg temperature, which will cooldown and depressurize the affected SG.

Technical Reference(s):	OP 3272 (Rev 8-12), Att 4, sheets 2-5	(Attach if not previously provided)
	OP 3272 (Rev 8-12), page 6 of 53	
	ES-0.1 (Rev 25-0) Foldout page	
	FR-H.2 (Rev. 9-0), step 7.a.RNO	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-07481 Given a set of plant conditions, determine the required actions to be taken per FR-H.2.	(As available)

Question Source: New

Question History:

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: High Containment Radiation		Group #		2
Ability to diagnose and recognize trends		K/A #	W/E16.GEN.2.4.47	
utilizing reference material		Importance Rating		4.2

Proposed Question:

With the plant initially at 100% power with Containment High-Range Monitor 3RMS-RE04A out of service, the following sequence of events occurs:

Time	Event
0 Minutes:	A locked rotor occurs on the "A" RCP, resulting in a reactor trip.
5 Minutes:	While in ES-0.1, *Reactor Trip Response*, the STA reports Fuel Drop Monitors 3RMS- RE 41 and 42 are reading 0.5 R/hr and trending up.
10 Minutes:	A Yellow Path is received on CTMT Radiation.
120 Minutes:	3RMS- RE 41/42 are reading 150 R/hr and continuing to trend up.
120 Minutes:	Containment High-Range Monitor 3RMS- RE 05A is reading 150 R/hr and trending up.
145 Minutes:	3RMS- RE 05A and 3RMS- RE 41/42 all read 175 R/hr and stable.
149 Minutes:	RMT1 reports the Core Damage Estimate is NOT yet complete.
150 Minutes:	A fault occurs on the "C" Steam Generator inside Containment, and the "C" SG rapidly depressurizes.
151 Minutes:	3RMS-RE-05A rapidly increases to 1000 R/hr.
151 Minutes:	3RMS-RE 41 and 42 readings have remained stable at 175 R/hr.

Using the Millstone 3 EAL Tables, what is the highest classification required for this event?

a) General Emergency – Alpha

b) Site Area Emergency – Charlie 2

c) Alert – Charlie 1

d) Unusual Event – Delta 1

Proposed Answer: C

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions related to radiation readings and make an emergency classification, which is a duty reserved for a SRO licensed individual. The Alert classification is first met at 10 minutes for "In-plant Radiation" (RA2), Radiation reading >5R/hr in Areas Requiring Access for Safe Shutdown (CTMT), since the Yellow path for CTMT Radiation is 10 R/hr, and for Loss of the Clad Barrier (FCB3) with sustained valid RE-04A/05A reading >5R/hr without RCS release inside CTMT. At 121 minutes, the classification remains Alert, C-1, even though CTMT radiation exceeds the Table 1 "2 to 4" hour limit of 125 R/hr, since this only applies with an RCS leak inside CTMT. "C" is correct, and "A", "B", and D" wrong, since at 151 minutes, the classification does not escalate, since the CTMT loss threshold is still not exceeded. "A" is plausible, since RE05A has rapidly increased above the RG1 "In-Plant Radiation" General Emergency threshold, but this is not valid, since the steam break inside CTMT is causing Temperature-Induced Current (TIC) on RE05A, and its readings are not to be considered valid until the effects of TIC have dissipated. Also, radiation impacts all three barriers are the barrier failure reference table. "B" is plausible, since this would be correct if the applicant uses the TIC-affected reading of RE05A to assess the CNB5 CTMT loss threshold, making two barriers lost, or, if the applicant views the RCS barrier as potentially lost due to uncontrolled pressure decrease (due to the steam break) and increasing containment radiation monitors (RCB3), which are elevated, but the rad monitor indication was stable during the steam break. "D" is plausible, since a steam break inside CTMT is not classifiable if Radiation levels were lower, if only the CTMT barrier is lost, Unusual Event would be correct, and unexpected RMS reading increasing by >1000 times normal readings is an Unusual Event (RU1).

Technical Reference(s):	MP-26-EPI-FAP06-003 (Rev. 8-0) EAL Tables	(Attach if not previously provided)

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

Learning Objective:	EP-00171 Given a plant condition and associated alarms and/or indications, classify an emergency event to include NRC classification and state posture code	(As available)
Question Source:	New	
Question History:		
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 # 86	Tier #		2
K/A Statement: Component Cooling Water:	Group #		1
Predict impact and mitigate: Loss of CCW pump	K/A #	008.A2.01	
	Importance Rating		3.6

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 aligned to its normal train, per OP 3330A, *Reactor Plant Component Cooling Water.*

The following sequence of events occurs:

1. The Work Control SRO reports the following:
   - The "C" RPCCW Pump Suction and Discharge Valves were inadvertently missed on the most recent RPCCW Valve Lineup Surveillance, which is required every 31 days.
   - The last time these valves were checked was 40 days ago.
2. A PEO is dispatched and completes the Valve Lineup on the "C" RPCCW Pump.
3. The "B" RPCCW Pump trips.
4. The crew enters AOP 3561, *Loss of Reactor Plant Component Cooling Water.*

Correctly complete the following statement regarding the initial requirement to log into a LCO 3.7.3, "Reactor Plant Component Cooling Water" ACTION due to the missed surveillance, and the required actions per AOP 3561 due to the "B" RPCCW Pump trip.

The crew__(1)___initially required to log into LCO 3.7.3 ACTION prior to completing the valve lineup on the "C" RPCCW Pump, and the crew is required to___(2)___ per AOP 3561 to restore cooling to "B" Train RPCCW loads.

a) (1) WAS NOT
   (2) cross-connect the RPCCW Containment Headers, and then shift from the "B" to the "C" RPCCW Pump and Heat Exchanger

b) (1) WAS
   (2) cross-connect the RPCCW Containment Headers, and then shift from the "B" to the "C" RPCCW Pump and Heat Exchanger

c) (1) WAS NOT
   (2) start the "C" RPCCW Pump, and complete shifting from the "B" to the "C" RPCCW Pump and Heat Exchanger

d) (1) WAS
   (2) start the "C" RPCCW Pump, and complete shifting from the "B" to the "C" RPCCW Pump and Heat Exchanger

Proposed Answer: C

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and select a procedure section based on the event in progress. Then apply Tech Spec Section 3.0/4.0 to current plant conditions. The "C" swing RPCCW Pump is normally aligned to the "B" Train. AOP 3561 will attempt to start the "C" RPCCW Pump if it is aligned to the affected train, and if not, will direct the crew to Attachment "A", which will cross-connect the RPCCW Containment Headers ("A" and "B" wrong, but plausible). "C" is correct, and "D" wrong, since per Surveillance Requirement 4.0.3, if it is discovered that Surveillance was not performed within its required interval, the requirement to enter the LCO may be delayed from the time of discovery, up to 24 hours or up to the limit of the surveillance interval, whichever is longer. This delay is permitted to allow performance of the Surveillance. "D" is plausible, since per 4.0.2, each Surveillance Requirement shall be performed within the specified time interval with a maximum allowable extension onto to exceed 25% of the surveillance interval, and 31 days x 1.25 is 38.75 days, which has been exceeded.

Technical Reference(s):	AOP 3561 (Rev 11-2), step 1	(Attach if not previously provided)
	AOP 3561 (Rev 11-2), Att A, steps 2 and 6	
	Tech Spec (Amend 241) Surveillance Req 4.0.2 and 4.0.3	
	Tech Spec (Amend 206) Surveillance Req 4.7.1.a	
	SP 3630A.3-001 (Rev 6-3), Page 4 of 4	

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07543 Given a set of plant conditions, determine the required actions to be taken per AOP 3561. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 87	Tier #		2
K/A Statement: Reactor Protection:	Group #		1
Predict impact and mitigate: Erratic power supply operation	K/A #	012.A2.04	
	Importance Rating		3.2

Proposed Question:

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

1. The SSPS A TROUBLE annunciator comes in and clears on MB4C.
2. The BOP operator reports VIAC 3 voltage is erratically spiking down on MB8.
3. The US starts looking at AOP 3564, *Loss of One Protective System Channel*, in case VIAC 3 completely deenergizes, or is required to be deenergized manually.
4. The BOP operator reports VIAC 3 voltage is continuing to erratically spike down.

Correctly complete the following statement to describe the status of RPS power supplies and which procedure will be utilized to determine which bistables will be tripped.

With VIAC 3 voltage spiking downward, the voltages being supplied to "A" Train RPS circuits by its 48V and 15V power supplies___(1)____.
If VIAC 3 deenergizes, the US will direct the appropriate RPS bistables to be tripped per___(2)____.

	(1)	(2)
a)	remain steady	AOP 3564, *Loss of One Protective System Channel*
b)	remain steady	AOP 3571, *Instrument Failure Response*
c)	spike downward	AOP 3564, *Loss of One Protective System Channel*
d)	spike downward	AOP 3571, *Instrument Failure Response*

Proposed Answer: B

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and select a specific Abnormal Operating Procedure based on based on the event in progress. It also requires the applicant to know the content of procedures to select a required course of action. The "A" Train of RPS receives electrical power from 120 VAC from two sources; VIAC 1 and VIAC 3. The slave relays receive power from VIAC 1 only ("C" and "D" plausible), and both provide input via an auctioneered high circuit to the 48 V and 15 V power supplies ("C" and "D" wrong). "A" is wrong, and "B" correct, since AOP 3564 directs the crew to AOP 3571 to address tripping of Bistables. "A" is plausible, since the crew will enter AOP 3564 on loss of VIAC.

Technical Reference(s):	EE-1BF (Rev. 32)	(Attach if not previously provided)
	SSPS Power Distribution Dwg	
	AOP 3564 (Rev. 10-0), step 10. RNO	
	AOP 3571 (Rev. 9-7), Att. A, page 6	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-05497 Describe the operation of the RPS under the following… Loss of Power…	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 88	Tier #		2
K/A Statement: Main and Reheat Steam: Predict	Group #		1
impact and mitigate: Steam dump malfunction	K/A #	039.A2.04	
Proposed Question:	Importance Rating		3.7

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

1. The MAIN STEAM RELIEF VV NOT CLOSED Annunciator is received on MB5.
2. The BOP operator reports that Atmospheric Relief Valve 3MSS*PV20D has spuriously opened.
3. The BOP operator reports that the cause of the failure is Steam Generator Pressure Transmitter 3MSS*PT20D has failed high.
4. The crew enters AOP 3571, *Instrument Failure Response.*
5. The RO reports Calorimetric 4 minute average is 3675 MWth.
6. The BOP operator manually closes 3MSS*PV20D at US direction.

Correctly complete the following statement to describe the required downpower rate and the bistable tripping requirement.

The exact wording of OP 3204, *At Power Operation* describing the required downpower rate is___(1)____ power to ≤ 100% power.
Technical Specification ACTION___(2)____require tripping bistables within 6 hours.

	(1)	(2)
a)	"immediately reduce"	DOES
b)	"immediately reduce"	DOES NOT
c)	"promptly reduce"	DOES
d)	"promptly reduce"	DOES NOT

Proposed Answer: D

Explanation (Optional): Question is considered a KA match since the KA catalogue describes the associated task as "Dump steam through the atmospheric relief/dump valves" and specific paths in the K/A's as "Atmospheric relief dump valves" (K/A 039.K1.02) and "Condenser steam dump" (K/A 039.K1.06). This question considers either path as a steam dump path. Question is considered SRO level since it requires the applicant to determine administrative requirements required to comply with the maximum power level limit in the Millstone 3 license. This question also requires the applicant to have detailed knowledge of which instruments have bistables associated with Tech Spec LCO's 3.3-1 and 3.3-2, since steam pressure does feed into ESF actuation signals. AOP 3571 directs the crew to manually close the failed-open relief valve, and notifies the crew that there are no Technical Specifications or bistables associated with the failed instrument ("A" and "C" wrong). "A" and "C" are plausible, since a failed steam generator pressure detector requires entry into AOP 3571 and requires the tripping of Bistables. OP 3204 actions need to be taken to address the overpower event caused by the malfunctioning steam relief valve. "D" is correct, since power has exceeded 100.5% (3,668 MWth), but not 102%. "B" is wrong, but plausible, since with power above 102% (3723 MWth), the crew is required to "immediately" reduce power to less than or equal to 100%, and notify RE.

Technical Reference(s): AOP 3571 (Rev. 9-7), Att. I (Attach if not previously provided)
OP 3204 (Rev. 19-0), section 1.2, page 3 of 78
OP 3204 (Rev. 19-0), section 4.3.1.a - d

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07497 Given a set of plant conditions, determine the required actions to be taken per OP 3204. (As available)

Question Source: Bank 85279

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.1, 43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 89	Tier #		2
K/A Statement: Service Water: Predict impact	Group #		1
and mitigate: Service water header pressure	K/A #	076.A2.02	
Proposed Question:	Importance Rating		3.1

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

1. The SERVICE WTR PUMP DIS PRES LO (MB1C, 4-3) annunciator is received.
2. The US enters the associated Annunciator Response Procedure (ARP).
3. The RO reports "A" Train Service Water Supply Pressure (3SWP-PI26A) has dropped to 15 psig and stabilized.
4. The RO reports the "A", "B", and "C" Service Water Pumps are running.
5. The RO reports Service Water Trains "A" and "B" TPCCW Supply Valves 3SWP*MOV71A and B have automatically closed.
6. A PEO reports large amounts of water spraying in the "A" Train Service Water Cubicle.
7. The crew takes action per OP 3353.MB1C, 4-3, and enters AOP 3560, *Loss of Service Water*.

Which of the choices below correctly completes the statement describing required procedural actions to be taken by the US?

Per AOP 3560,___(1)____, then go to/remain in___(2)____to continue mitigating the event.

a) (1) open 3SWP*MOV71B
   (2) AOP 3561, *Loss of Reactor Plant Component Cooling Water*

b) (1) open 3SWP*MOV71B
   (2) AOP 3560, *Loss of Service Water*

c) (1) verify 3SWP*MOV71B closed
   (2) AOP 3579, *Response to Turbine Runback/Loss of Turbine Load*

d) (1) verify 3SWP*MOV71B closed
   (2) E-0, *Reactor Trip or Safety Injection* and AOP 3560 in parallel

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and select Abnormal Operating Procedures based on decision points in the AOP for the specific event. Service Water System low pressure automatically starts the second pump in the affected train, and closes the SWP supply path to TPCCW. OP 3353.MB1C, 4-3 directs the crew to start the second SWP Pump in the unaffected train, and place the affected train's pumps in Pull-To-Lock to address the header rupture. Then AOP 3560 will direct the crew to open AOV71B to restore cooling to TPCCW (to prevent a plant runback on high stator cooling temperature), and then transition to AOP 3561, to restore cooling to the RCPs, since RPCCW has lost SWP cooling on the affected train ("A" is correct and "B" wrong). "B" is plausible, since AOP 3560 takes further actions for a loss of service water if the crew is able to restore SWP flow to the affected SWP train. "C" and "D" are wrong, since AOP 3560 assumes the ARP has addressed the pipe break, and AOP 3560 restores cooling to TPCCW. "C" and "D" are plausible, since most AOPs do not assume ARP action has been taken, and if TPCCW cooling is not restored, a stator cooling runback will occur.

Technical Reference(s): OP 3353.MB1C (Rev. 6-3), 4-3 (Attach if not previously provided)
AOP 3560 (Rev. 8-1), steps 1 and 2
P&ID 133A (Rev. 44), B (Rev. 86), and D (Rev. 44)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07541 Given a set of plant conditions, determine the required actions to be taken per AOP 3560 (As available)

Question Source: New

Question History:

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 # 90	Tier #		2
K/A Statement: Containment:	Group #		1
Ability to perform specific system and integrated	K/A #	103.GEN.2.1.23	
plant procedures during all modes of operation	Importance Rating		4.4

Proposed Question:

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

1. The Containment Unidentified Leakage Sump Pump (3DAS-P10) is tagged out.
2. The "A" Containment Drains Sump Pump (3DAS-P2A) is off.
3. The "B" Containment Drains Sump Pump (3DAS-P2B) is off.
4. Containment Atmosphere Particulate Monitor (3CMS-RE22) is in service.
5. Containment Atmosphere Gaseous Monitor (3CMS-RE22) is tagged out.
6. The "A" CAR Fan (3HVU-FN1A) is running.
7. The "B" CAR Fan (3HVU-FN1B) is off.
8. The "C" CAR Fan (3HVU-FN1C) is tagged out.
9. The Plant Process Computer and all associated computer points are functioning.

The "A" CAR Fan trips.

Which of the following actions are required to be taken to ensure the minimum required equipment is available to meet RCS Leak Detection Systems OPERABILITY requirements prior to entering MODE 4?

a) Start the "B" CAR Fan; and place AT LEAST one Containment Drains Sump Pump in Automatic.

b) Start the "B" CAR Fan; and place BOTH Containment Drains Sump Pumps in Automatic.

c) Restore the Containment Atmosphere Gaseous Monitor to service; and place AT LEAST one Containment Drains Sump Pump in Automatic.

d) Restore the Containment Atmosphere Gaseous Monitor to service; and place BOTH Containment Drains Sump Pumps in Automatic.

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to apply Tech Spec Basis knowledge to determine operability. "A" is correct, since starting a CAR fan is required to ensure a representative sample is being reaching the Radiation Monitor, and restoring the "A" Containment Drains Sump Pump meets the leak detection requirement of either the Unidentified Leakage Sump Pump OR either Containment Drains Sump Pump. Tech Spec LCO 3.4.6.1 requires the Containment Atmosphere Particulate Monitor, but not the Gaseous Monitor ("C" and "D" wrong, but plausible). "B" is wrong, but plausible, since the Containment Unidentified Leakage Sump Pump can be used to meet leak detection requirements, but this function can also be met by either Containment Drains Sump Pump.

Technical Reference(s):	OP3312B (Rev. 7-2), Prerequisite 2.1.2	(Attach if not previously provided)
	OP 3335B (Rev. 14-1), sections 4.1.2 and 4.1.6	
	OP 3302A (Rev. 15-7), Precaution 3.1	
	Tech Spec LCO 3.4.6.1 (Amend 244) and Basis (LBDCR No. 07-M3-032 and 11-M3-004)	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04249 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…	(As available)

Question Source: New

Question History:

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: Hydrogen Recombiner and	Group #		2
Purge Control: Knowledge of conditions	K/A #	017.GEN.2.2.38	
and limits in the facility license	Importance Rating		4.5

Proposed Question:

Initial conditions:

- A LOCA is in progress.
- The crew is performing actions per FR-C.1, *Response to Inadequate Core Cooling.*

For which condition will FR-C.1 direct the crew to "Consult ADTS to determine if Ctmt purge system should be placed in service", and what FSAR assumption is made about an event requiring the use of Containment Hydrogen Monitors?

a) Containment hydrogen concentration has reached 5%. The event in progress is assumed to be a design basis LOCA.

b) Containment hydrogen concentration has reached 5%. The event in progress is assumed to be beyond design basis.

c) It has been 24 hours since event initiation. The event in progress is assumed to be a design basis LOCA.

d) It has been 24 hours since event initiation. The event in progress is assumed to be beyond design basis.

Proposed Answer: B

Explanation (Optional): Question is considered SRO level since it requires detailed knowledge about an FR-C.1 step beyond system knowledge or overall procedure strategy, and also knowledge of FSAR assumptions. Question is considered a KA match since the facility license states "The facility is... described in the licensees' 'Final Safety Analysis Report...", and other KA statements already cover information included in Technical Specifications, but not FSAR assumptions. SRO knowledge of the FSAR assumptions supports requirements of the operating license. FR-C.1 requires operators to startup the hydrogen monitors to assess the degree of core damage during beyond design basis accidents ("A" and "C" wrong), and if an explosive gas mixture threatens Containment integrity during beyond design basis events, then containment purge system can be used to lower hydrogen concentration to protect the integrity of the containment boundary ("B" correct, "D" wrong). "A", "C", and "D" are plausible, since much of the FSAR assumes design basis events, and numerous operator credited actions in the FSAR (such as time to place th hydrogen monitors in service) are driven by time.

Technical Reference(s): FR-C.1 (Rev 17-0), step 6 (Attach if not
FSAR (Rev 24.3) Section 6.2.5, pages 6.2.81 and 83 previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04261 Describe the major administrative or procedural precautions and limitations placed on the operation of the Containment Ventilation System, and the basis for each. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 92	Tier #		2
K/A Statement: Spent Fuel Pool Cooling: Ability to explain and apply system limits and precautions	Group #		2
	K/A #	033.GEN.2.1.32	
	Importance Rating		4.0

Proposed Question:

Initial Conditions:

- The plant is at 100% power.
- A Spent Fuel Assembly is being moved to a new location in the Spent Fuel Pool.
- All precautions and limits of OP 3305, *Fuel Pool Cooling and Purification*, have been verified to be met.

A transient occurs in the Spent Fuel Pool, and Fuel Pool parameters stabilize at the following:

- Fuel Pool temperature: 152°F
- Fuel Pool level: 36%

Select the choice which correctly identifies which limit(s) has/have been exceeded, and which procedure the US is required to enter.

a) Fuel Pool temperature has exceeded the FSAR Fuel Pool long-term design temperature limit, but is still within the maximum peak temperature limit. The US will enter OP 3353.MB1A, 4-4, *Fuel Pool Temp Hi.*

b) Fuel Pool temperature has exceeded both the FSAR long-term design temperature limit and the maximum peak temperature limit. The US will enter OP 3353.MB1A, 4-4, *Fuel Pool Temp Hi.*

c) Fuel Pool level has dropped below the level necessary to ensure 99% of the iodine that would be released by a fuel assembly would be filtered. The US will enter OP 3353.MB1A, 3-4, *Fuel Pool Level Lo.*

d) Fuel Pool level has dropped below the level necessary to ensure dose rate at the surface of the Fuel Pool remains less than 2.5 mrem per hour. The US will enter OP 3353.MB1A, 3-4, *Fuel Pool Level Lo.*

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to have detailed knowledge of FSAR design information, and Tech Spec bases; as well as determine entry conditions for annunciator response procedures. "A" is correct, since Fuel Pool temperature has exceeded the long-term design limit of 150°F listed in FSAR 9.1.3.1, but not the maximum peak temperature limit of 200°F ("B" wrong). "B" is plausible, since temperature has exceeded the entry requirement for OP 3353.MB1A, 4-4 of 125°F, and the long term temperature limit. "C" and "D" are wrong, since fuel pool level is above the low-level entry condition for OP 3353.MB1A, 3-4 of 35%. "C" and "D" are plausible, since level is at the low end of the normal range, and the consequences listed are bases for minimum fuel pool level above the fuel in the storage racks, and the minimum level above the fuel rod being moved.

Technical Reference(s): OP 3210B (Rev. 10-3), Precaution 3.14 (Attach if not previously provided)

OP 3305 (Rev. 21-7), Section 1.2, Precaution 3.1, and Att. 1

Tech Spec Bases for 3 / 4.9.11 (LBDCR No. 06-M3-026)

FSAR 9.1.3.1 (Rev. 22-3), item 14

OP 3353.MB1A (Rev. 3-2), 3-4

OP 3353.MB1A (Rev. 3-2), 4-4

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-056542 Describe the major Administrative or Procedural Precautions and Limitations placed on the operation of the Spent Fuel Pool Cooling System and the basis for each. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.1, 43.2, 43.5, and 43.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 93	Tier #		2
K/A Statement: Liquid Radwaste:	Group #		2
Ability to apply Technical Specifications for a system	K/A #	068.GEN.2.2.40	
Proposed Question:	Importance Rating		4.7

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

1. A discharge of the "A" Low Level Waste Drain Tank (LLWDT) is commenced.
2. It is discovered that liquid waste radiation monitor 3LWS-RE70 is no longer functioning.
3. The crew terminates the discharge.

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

Which of the below choices describes required actions prior to recommencing the discharge?

a) Direct I&C to install a temporary monitor with an alarm setpoint below that of the 3LWS-RE70 setpoint.

b) Initiate efforts to repair the instrument. Perform at least two independent samples, independent release calculations, and independent discharge valve lineups.

c) Recirculate the LLWDT an additional 15 minutes, and perform two independent discharge valve lineups.

d) Reconfirm release calculations. Direct Chemistry to take samples every 15 minutes while the discharge is in progress to ensure effluent is within limits.

Proposed Answer: B

Explanation (Optional): Question is considered SRO level since it requires the applicant to apply REMODCM administrative requirements related to liquid radioactive release approvals. "B" is correct, and "A", "C", and "D" wrong, since REMODCM Table V.C-1 ACTION A requires best efforts to repair the instrument; and independent samples, release calculations, and discharge valve lineups prior to initiating a release. "A" and "D" are plausible, since numerous actions with inoperable rad monitors or other discharge monitors involve temporary monitors or manual samples. "C" is plausible, since recirculating the tank is required prior to its discharge.

Technical Reference(s): MP-22-REC-BAP01 (REMODCM) (Rev. 27-0), V.C.1, page 134; MP-22-REC-BAP01 (Rev 27-0), Table V.C.-1, pages 135 & 136 (Attach if not previously provided)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04868 Given a plant condition or equipment malfunction… evaluate tech spec applicability and determine required action… (As available)

Question Source: Bank #74490

Question History: Millstone 3 2009 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2, 43.4, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 94	Tier #		3
K/A Statement: Generic:	Group #		1
Ability to perform specific system and integrated	K/A #	GEN.2.1.23	
plant procedures during all modes of operation	Importance Rating		4.4

Proposed Question:

The Plant is in MODE 3 with a cooldown in progress in accordance with OP 3208, *Plant Cooldown.*

Due to a Tech Spec ACTION requirement, the crew is attempting to cooldown at the maximum rate allowed by the plant cooldown goal.

The following data has been logged over the last hour:

TIME	RCS TEMP	RCS PRESS
1500	550.0°F	2075 psia
1515	531.5°F	1700 psia
1530	513.0°F	1500 psia
1545	493.5°F	1350 psia
1600	474.0°F	1125 psia

What action, if any, is required to be taken with the cooldown at Time 1600?

a) Increase the cooldown rate, since it is below the plant cooldown goal.

b) Maintain current cooldown rate, since it is at the plant cooldown goal.

c) Decrease the cooldown rate, since it exceeds the plant cooldown goal, but not the Tech Spec limit.

d) Stop the cooldown, since it exceeds both the plant cooldown goal and the Tech Spec limit.

Proposed Answer: C

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and to have detailed knowledge of Tech Spec Surveillance procedures to select a required course of action based on administrative limits and Tech Spec limits, based on a cooldown rate that is required to be checked both over the last hour and over the last 15 minutes, and limits that change based on RCS temperature. The administrative limit is 70 +/- 5°F in any one-hour period, or 1.25°F per minute (18.75°F in 15 minutes). The Tech Spec limit is 80°F/hr, or 1.33°F per minute (20°F in 15 minutes). Cooldown over the last hour was 76°F, and ΔT for current 15 minutes is 19.5°F in 15 minutes, or 1.30 °F/min. Based on plant conditions the Tech Spec limit is NOT being exceeded ("D" wrong), but the cooldown rate should be decreased to approx. 1.25°F/min to comply with the Admin limit ("C" correct, "A" and "B" wrong).

Technical Reference(s): SP 3601G (Rev. 9-0), step 4.2.3.d & e (Attach if not previously provided)
Form 3601G.2-001 (Rev. 5-4), pages 2 - 4
OP 3208 (Rev. 22-4), step 4.2.9.e

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07503 Given a set of plant conditions, determine the required actions to be taken per OP 3208. (As available)

Question Source: Bank 78941

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 95	Tier #		3
K/A Statement: Generic	Group #		2
Knowledge of	K/A #	GEN.2.1.35	
fuel handling responsibilities of SROs	Importance Rating		3.9

Proposed Question:

Current Conditions:

- The plant is in MODE 6.
- Westinghouse Source Range Channel N31 is out of service for surveillance testing.
- Both Gammametrics Channels are OPERABLE.
- Both Fuel Building Exhaust Filter Trains are available.
- The "A" Train Fuel Bldg Filter is running, and the "B" Train is not running.
- The Containment Equipment Hatch is open under administrative control, and is capable of being closed and bolted in place.
- Five hours ago, RHR flow was suspended for one hour to improve fuel pool visibility.
- RHR flow is currently 2500 gpm.

The refueling team has established communications with the control room, and has requested permission from the Refueling SRO to move the next fuel bundle from the fuel building to the core.

Which of the above conditions prevents the Refueling SRO from recommencing fuel movement?

a) Westinghouse SR Channel N31 is out of service.

b) The number of running Fuel Building Filters is insufficient.

c) The Containment Equipment Hatch is open.

d) Residual Heat Removal System flow is insufficient.

Proposed Answer: D

Explanation (Optional): Question is considered SRO level since it requires the applicant to know administrative requirements associated with refueling activities. "A" is wrong, since Gammametrics detectors can suffice for Source Range Counts. "A" is plausible, since the normal method of monitoring counts during refueling is Westinghouse SR NIS. "B" and "C" are wrong, but plausible, since one train of Fuel Bldg filters is run during refueling, and the Containment Equipment Hatch is allowed to be open under administrative control. Tech Spec 3.9.12, which limited fuel building boundary breaches in MODE 6 under administrative control, and contained Fuel Building Filter requirements, has been deleted. "D" is correct, since RHR flow is allowed to be suspended for a maximum of one hour out of the previous 8 hours (this has been exceeded). Otherwise, 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 are less than optimal.

Technical Reference(s):	OP 3210B (Rev. 10-3), Prerequisites, especially 2.1.12 and 18	(Attach if not previously provided)
	OP 3210B (Rev. 10-3), Precautions, especially 3.9 and 3.10	
	SP3613F.3-001 (Rev 6-1), page 2 of 2	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-06495 Describe the Stop Work requirements with regards to fuel movement in accordance with EN 31007, Refueling Operations.	(As available)

Question Source: Bank 78793

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2, 43.4, 43.5, 43.6, and 43.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 96	Tier #		3
K/A Statement: Generic:	Group #		2
Ability to determine operability and/or	K/A #	GEN.2.2.37	
availability of safety related equipment	Importance Rating		4.6

Proposed Question:

Initial Conditions:

- The plant is in MODE 0.
- An "A" Train Electrical Outage is in progress.
- The "B" Spent Fuel Pool Cooling (SFC) Pump is in operation.
- An Electrician, carrying a beeper, is available to establish temporary power to the "A" SFC Pump.
- The "B" and "C" RPCCW Pumps are available.
- The "B" and "D" Service Water Pumps are available.

The "B" SFC Pump trips.

Is the "A" SFC Pump allowed to be credited when determining Shutdown Risk (SDR) Condition Colors; and can it be considered available for use in EOP 3505A, *Loss of Spent Fuel Pool Cooling*?

	Credit for SDR	Available for use in EOP 3505A
a)	Yes	Yes
b)	Yes	No
c)	No	Yes
d)	No	No

Proposed Answer: C

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and determine whether Fuel Pool Cooling Pump administrative requirements are adequate to meet both Shutdown Risk administrative requirements during refueling and also whether it complies with availability requirements for a 3500 series EOP. Per Tech Spec Basis 3/4.9.1, MODE 0 is defined as "the Operational MODE where all fuel assemblies have been removed from containment to the Spent Fuel Pool." The "B" SFC Pump is available for EOP 3505A, since it does have its administrative requirements (electrician, 2 CCP pumps, and 2 SWP pumps) met ("B" and "D" wrong), but it is not credited for SDR, since it does not have its normal power supply available ("A" wrong, "C" correct). "A" is plausible, since the "A" SFC Pump does have its admin requirements met to use backup power. "B" and "D" are plausible, since normal power is not available, and there are several admin requirements needed to credit the pump.

Technical Reference(s):	OU-M3-201 (Rev.6-0), Att. 2, Page 4	(Attach if not previously provided)	
	EOP 3505A (Rev. 10-0), Att E		

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-07622 Given a specific set of plant conditions, complete a shut down safety assessment check list using OU-M3-201, Shutdown Safety Assessment Checklist	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5 and 43.6

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 97	Tier #		3
K/A Statement: Generic:	Group #		2
Knowledge of the process used to track	K/A #	GEN.2.2.43	
inoperable alarms	Importance Rating		3.3

Proposed Question:

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

1. The RO reports one annunciator fails to illuminate during the daily main board annunciator test.
2. I&C determines the annunciator is disabled, and a blue dot is placed on the annunciator window.

What administrative classification is required to be assigned to this annunciator; and what responsibility is specifically assigned to the Shift Manager in response to the disabled annunciator?

a) This is classified as an Operator Burden. The SM is responsible for determining the Reactivity Management Level classification, if required.

b) This is classified as an Operator Burden. The SM is responsible for ensuring the annunciator is identified as a contributor to the Ops Aggregate Impact.

c) This is classified as a Control Room Deficiency. The SM is responsible for determining the Reactivity Management Level classification, if required.

d) This is classified as a Control Room Deficiency. The SM is responsible for ensuring the annunciator is identified as a contributor to the Ops Aggregate Impact.

Proposed Answer: D

Explanation (Optional): Question is considered SRO level since it requires the applicant to know administrative processes specifically assigned to an SRO licensed individual for a disabled annunciator. A disabled annunciator is to be classified as a "Control Room Deficiency" and would only be considered an "Operator Burden" if it was to be established that Operators would be required to take significant compensatory actions during normal operations ("A" and "B" wrong, but plausible). The classification of this event is significant since it determines the point value to be assessed against the Ops Aggregate Impact (Operator Burden = 2 points, and Control Room Deficiency = 1 point). "C" is wrong, but plausible, since the Reactivity Management Level determination is the responsibility of the Reactivity Management Team. "D" is correct, since the SM responsibility is to ensure the annunciator is identified as a contributor to the Ops Aggregate Impact.

Technical Reference(s):	OP-AA-100 (Rev. 25-0), Att 6, Item 7	(Attach if not previously provided)
	OP-AA-1700 (Rev. 6-0), Sections 5.2.1, 5.3.3, 5.3.5	
	OP 3353 (Rev. 8-3), Section 4.3	
	OPSTAT Database Instructions (4/1/09), Item 2, Note 3	

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-01956 Demonstrate compliance with Conduct of Operations, OP-AA-100 (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.3 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 98	Tier #		3
K/A Statement: Generic: Ability to use radiation	Group #		3
monitoring systems, such as fixed monitors, portable	K/A #	GEN.2.3.5	
survey instruments, personnel monitoring equipment, etc.	Importance Rating		2.9

Proposed Question:

A LOCA occurs, and Initial Conditions are as follows:

- An ALERT-Charlie 1 has been declared at Millstone 3.
- The CR-DSEO is preparing the Incident Report Form.
- The CR-DSEO is currently determining if the "Radiological Release in progress due to event" box is required to be checked on the IRF form.

The CR-DSEO gathers the following information from the Radiation Monitor Computer:

1. ESF Bldg Normal Vent Exhaust Monitor 3HVQ-RE49 shows an increasing trend, but is NOT in ALERT.
2. Turbine Building Stack Monitor 3HVR-RE10A is in ALERT, but is NOT in ALARM.
3. Liquid Waste Effluent Monitor 3LWS-RE70 is in ALERT, but is NOT in ALARM.
4. Containment Recirc Cooler Discharge Monitor 3SWP-RE60A is in ALARM.

In accordance with MP-26-EPI-FAP06, Classification and PARs, which one of these Radiation Monitors specifically requires the CR-DSEO to select the "Radiological release in progress due to event" Box?

a) 3HVQ-RE49 increasing trend.

b) 3HVR-RE10A in ALERT

c) 3LWS-RE70 in ALERT

d) 3SWP-RE60A in ALARM

Proposed Answer: B

Explanation (Optional): 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 Radiation Monitor input to checking the block "Radiological release in progress due to event" is a gaseous effluent radiation monitor ("A" plausible) in ALERT ("B" correct, "A" wrong) or ALARM ("C" and "D" plausible). "C" and "D" are wrong, since these are not gaseous effluent monitors. "A" is plausible, since it is a gaseous effluent monitor, and an increasing trend is considered "not normal" when diagnosing RMS Alarm/Alert status if it is anticipated it will reach the ALERT setpoint. "C" and "D" are plausible, since both of these monitors are monitoring an effluent point, and RE10A is only in ALERT.

Technical Reference(s):	MP-26-EPI-FAP06 (Rev. 8-0), Section 2.1.8	(Attach if not previously
	MP-26-EPI-FAP06 (Rev. 8-0), Att. 6	provided)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-02534 The Shift Manager and Unit Supervisor will perform all administrative actions necessary to protect the public in accordance with emergency plan procedures.	(As available)

Question Source:	New
Question History:	
Question Cognitive Level:	Memory or Fundamental Knowledge
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: Generic:	Group #		4
Knowledge of fire protection procedures	K/A #	GEN.2.4.25	
Proposed Question:	Importance Rating		3.7

With the plant initially at 100% power with the Fire Protection CO_2 System in its normal lineup, the following sequence of events occurs:

1. It is discovered that the pressure switch has failed that supplies the electrical auto-close signal to Electro-Thermal Link (ETL) fire damper (3HVR*DMPF142) between the East and West MCC/Rod Control Area.
2. The crew is preparing to lockout CO_2 to the East and West MCC/Rod Control Areas per OP 3341C *Carbon Dioxide Fire Protection System* to allow repairs.

What action is required with the East and West MCC/Rod Control Area Lockout Ball Valves to lock out CO_2 to the areas; and using **TRM 3.7.12.3, attached to this exam**, what fire watch requirement exists?

a) The normally closed Lockout Ball Valve needs to be locked. An hourly fire watch patrol is required for the East and West MCC/Rod Control Areas.

b) The normally closed Lockout Ball Valve needs to be locked. A continuous fire watch is required for the East and West MCC/Rod Control Areas.

c) The normally open Lockout Ball Valve needs to be closed and locked. An hourly fire watch patrol is required for the East and West MCC/Rod Control Areas.

d) The normally open Lockout Ball Valve needs to be closed and locked. A continuous fire watch is required for the East and West MCC/Rod Control Areas.

Proposed Answer: A

Explanation (Optional): Question is considered SRO level since it requires the applicant to apply Tech Spec/TRM actions based on specific plant conditions, as well as administration of fire protection program requirements. The US will direct the PEO to lock the normally closed lockout ball valves, since the MCC/RCA areas are manually actuated CO2 discharge areas ("C" and "D" wrong). "C" and "D" are plausible, since these areas were originally designed for automatic actuation, and are capable of being aligned for automatic actuation. Also, several areas are currently aligned for automatic operation. "A" is correct, and "B" wrong, since an hourly patrol is required for an inoperable CO2 area as long as its fire dampers are operable, and the fire damper will still close, even without the voltage signal, due to high temperature during a fire. "B" is plausible, since a continuous fire watch is required if the damper is inoperable, and there is a problem with its auto-close signal.

Technical Reference(s):	OP 3341C (Rev. 16-9), Section 4.23 TRM 3.7.12.3 (LBDCR 07-MP3-018)	(Attach if not previously provided)

Proposed references to be provided to applicants during examination: **TRM 3.7.12.3**

Learning Objective:	MC-04587 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…	(As available)

Question Source:	Bank 85282
Question History:	Millstone 3 2009 NRC Exam
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.43.1, 43.2, and 43.5
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 100	Tier #		3
K/A Statement: Generic: Knowledge of RO tasks	Group #		4
performed outside the control room during an	K/A #	GEN.2.4.34	
emergency, and resultant operational effects	Importance Rating		4.1

Proposed Question:

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

1. The crew enters EOP 3503, *Shutdown Outside Control Room.*
2. The extra Licensed Operator trips the Reactor Coolant Pumps at the Switchgear.
3. The crew is preparing to cooldown the RCS using EOP 3504, *Cooldown Outside Control Room.*
4. A PEO reports only the "A" CRDM cooling fan is available.

What action is the US required to direct?

a) The crew is required to go to ES-0.3, *Natural Circulation Cooldown With Steam Void in Vessel (with RVLMS)*, and conduct the cooldown with subcooling GREATER THAN 132°F.

b) The crew is required to go to ES-0.3, *Natural Circulation Cooldown With Steam Void in Vessel (with RVLMS)*, and conduct the cooldown at LESS THAN 50°F/hr.

c) The crew will remain in EOP 3504, align a reactor vessel head vent letdown path, and conduct the cooldown with subcooling GREATER THAN 132°F.

d) The crew will remain in EOP 3504, align a reactor vessel head vent letdown path, and conduct the cooldown at LESS THAN 50°F/hr.

Proposed Answer: D

Explanation (Optional): Question is considered SRO level since it requires the applicant to assess plant conditions and know the content of a 3500 series EOP to select a required course of action for the specific event. The extra licensed operator tripped the RCPs per EOP 3503, step 14. This action requires the crew to cooldown the plant at 50°F per hour, rather than the normal cooldown rate limit, to avoid allowing a steam bubble to form in the reactor vessel head. "A" and "B" are wrong, since ES-0.3 is used during natural circulation conditions when there is an urgency to reach Cold Shutdown conditions, and a bubble is intentionally allowed to grow in the head during the cooldown. "A" and "B" are plausible, since a natural circulation cooldown is in progress, and head cooling is limited without a second CRDM Cooling Fan. "D" is correct, since with one CRDM cooling fan available, the US will direct the crew to align the head vent path at the Aux Shutdown Panel to help cool the vessel head. "C" is wrong, but plausible, since the 132°F subcooling is only a requirement if the head vent path cannot be aligned during a cooldown in ES-0.2, *Natural Circulation Cooldown.*

Technical Reference(s): EOP 3503 (Rev. 15-3), step 14 (Attach if not previously provided)
EOP 3504 (Rev. 8-6), steps 4 and 6

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07511 Given a set of plant conditions, determine the required actions to be taken per EOP 3504. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments: