Examination Outline Cross-reference:	Level	RO	SRO
Question # 1	Tier #	1	1
Reactor Trip: Interrelations between	Group #	1	1
Reactor Trip and Reactor Trip Status Panel	K/A #	EPE.007.EK2.03	
	Importance Rating	3.5	3.6

Proposed Question:

With the plant at 7% power, the following sequence of events occurs:

1. Numerous annunciators are received.
2. The RO checks the bistable status panel on MB4.
3. The reactor trips.

Which bistable status light combination on MB4 would have directly initiated the reactor trip signal?

A. Four RCS "Loop Flow Low" bistable lights are lit.

B. Three "Pzr Level Hi" bistable lights are lit.

C. Two "Pzr Pres Hi" bistable lights are lit.

D. Two "Pzr Press Lo" bistable lights are lit.

Proposed Answer: C

Explanation (Optional): "C" is correct, since the coincidence for the Pzr High Pressure Reactor Trip is 2/4. "A", "B", and "D" are wrong, since these trips are blocked below P-7 (10% reactor power). "A", "B", and "D" are plausible, since each of these signals would produce an automatic reactor trip if the correct power level and coincidence were met.

Technical Reference(s): Functional Drawings 5 (Rev. K), 6 (Rev. H), and 7 (Rev. M).
(Attach if not previously provided)
(including version/revision number)

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: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 2	Tier #	1	1
Small Break LOCA:	Group #	1	1
Knowledge of system setpoints, interlocks and automatic actions associated with EOP entry conditions	K/A #	EPE.009.GEN.2.4.2	
	Importance Rating	4.5	4.6

Proposed Question:
Initial Conditions:

- The crew is performing a plant cooldown in accordance with OP 3208 *Plant Cooldown*.
- Pzr Pressure is 1950 psia.
- The crew has just verified RCS pressure is below P-11, and completed all associated ESF BLOCKS required by OP 3208.

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

1. Pzr pressure drops below 1900 psia.
2. Pzr pressure drops below 1892 psia.
3. CTMT pressure increases above 18 psia.
4. Steam pressure drops below 660 psig due to an operator controlled cooldown.

Assuming operators did not manually actuate SIS, when did an automatic Safety Injection actuate during this event?

A. When Pzr Pressure dropped below 1900 psia.

B. When Pzr pressure dropped below 1892 psia.

C. When CTMT pressure increased above 18 psia.

D. When steam pressure dropped below 660 psig.

Proposed Answer: C

Explanation (Optional): On the plant shutdown, the low pressure reactor trip (1900 psia) was blocked below P-7 ("A" wrong, but plausible). When the crew instated the blocks below P-11, the low pressure SI was blocked ("B" wrong, but plausible), and the low steamline pressure SI was blocked ("D" wrong, but plausible). Also, the low steamline pressure MSI was blocked, and the high steam pressure rate MSI was instated. The CTMT Hi 1 pressure SI is still in service, actuating SIS at 18 psia ("C" correct).

Technical Reference(s):	Functional Drawings 6 (Rev. H) and 8 (Rev. J)
(Attach if not previously provided)	E-0 (Rev. 025) Entry Conditions, pages 2 and 3
(including version/revision number)	OP 3208 (Rev. 021-03), Step 4.2.5

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… ESF Actuation Signals… Protective Interlocks… ESF Block and Reset Switches. (As available)

Question Source: New

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 # 3	Tier #	1	1
Large Break LOCA:	Group #	1	1
Ability to verify alarms are consistent with plant	K/A #	EPE.011.GEN.2.4.46	
conditions	Importance Rating	4.2	4.2

Proposed Question:

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

1. The RCS rapidly depressurizes to Containment atmospheric pressure.
2. CDA actuates.
3. The crew performs their shift brief at E-0 *Reactor Trip or Safety Injection*, step 16.
4. The US directs the board operators to walk down their boards and report any unexpected annunciators and indications prior to coming out of Master Silence.

During his walk-down, the RO observes the status of the following 4 annunciators/indications:

- The SAFEGUARDS AREA FLOODING annunciator (MB1C, 2-8) is dark.
- The Red Path Status Tree for "RCS Integrity" on SPDS is lit, with a tag directing entry into FR-P.1 *Response to Pressurized Thermal Shock*.
- The Orange Path for "Containment" on SPDS is lit, with a tag directing entry into FR-Z.1 *Response to High Containment Pressure*.
- The COLD LEG INJECTION PERM P-19 annunciator (MB4D, 4-5) is dark.

Which of these conditions is the RO required to report as **"unexpected"**?

A. The SAFEGUARDS AREA FLOODING annunciator, since it should be lit. Safeguards areas need to be filling with water in preparation for Cold Leg Recirculation.

B. The Red Path Status Tree for "RCS Integrity" on SPDS, since it should NOT be lit. The RCS is depressurized, and is not capable of repressurization.

C. The Orange Path for "Containment" on SPDS, since it should also tag entry into FR-Z.2 *Response to Containment Flooding*. CTMT sump level should be elevated.

D. The COLD LEG INJECTION PERM P-19 annunciator, since it should be lit. This ensures the proper ECCS valve lineup exists for a large break LOCA.

Proposed Answer: D

Explanation (Optional): "A" is wrong, since the areas monitored by the Safeguards Area Flooding annunciator are outside CTMT. "B" is wrong, since the RCS loop with the LOCA has experienced Blowdown, followed by cold ECCS water flowing into the loop and out the break, and the FR-P.1 status tree monitors for excessive cooldown in the loops. "C" is wrong, since the CTMT flooding setpoint is based on water in excess of DBA RCS/RWST water. "D" is correct, since P-19 (a new modification at Millstone 3) monitors RCS pressure, and only permits the cold leg injection valves to open if RCS pressure drops less than 1900 psia. This annunciator being dark indicates that P-19 has not allowed the cold leg injection valves to open. "A" and "C" are plausible, since large quantities of water have been released from the RCS. "B" is plausible, since the accidents RCS Integrity is concerned about are Cold Overpressure, and Pressurized Thermal Shock; and for a large break LOCA, the RCS cannot re-pressurize.

Technical Reference(s): OP 3208 (Rev 021-03), step 4.2.6

OP 3353.MB1C (Rev 005-13), 2-8,

(Attach if not previously provided) RCS Integrity Status Tree (Rev. 006)

(including version/revision number) WOG Basis Document (Rev. 2) for FR-Z.2

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04912 For a Large Break LOCA… Describe the symptoms of the event… (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.10 and 43.5

Comments:

Examination Outline Cross-reference:		Level	RO	SRO
Question # 4		Tier #	1	1
RCP Malfunctions:		Group #	1	1
Operational implications of the consequences of an		K/A #	APE.015/17.AK1.02	
RCPS failure		Importance Rating	3.7	4.1

Proposed Question:

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

1. The "A" Reactor Coolant Pump motor upper oil reservoir level starts slowly increasing.
2. The RCP A UPR OIL RSVR LVL HI annunciator is received on MB4.
3. No other abnormal annunciators are lit.

What is causing the increasing level, and will pump lubrication be maintained?

A. Cooling water is leaking into the reservoir. The pump will lose lubrication.

B. Cooling water has been isolated to the reservoir, and the oil is expanding as it heats up. The pump will lose lubrication.

C. The lube oil fill valve is leaking by. Pump lubrication will be maintained.

D. Upper radial bearing wear is reducing oil flow, backing up oil in the reservoir. Pump lubrication will be maintained.

Proposed Answer: A

Explanation (Optional): "A" is correct since RPCCW pressure is above oil pressure, so increasing oil reservoir level is indicative of a cooling water leak. OP3353.MB4B 4-2A directs the operators to check RPCCW surge tank for indications of in-leakage. "B" is wrong since no other annunciators are lit, and loss of cooling water would result in a and a RCP A Cooler Supply Pressure Lo annunciator (MB4B, 3-2B). "B" is plausible, since oil expands as it heats up. "C" is wrong since oil is added to the RCPs via drums that are not normally lined up to the RCP. "C" is plausible, since a manual oil makeup valve exists, and if it leaked by while an oil source was attached, level would increase. "D" is wrong since radial bearing wear will cause oil flow to increase due to increased mechanical clearances. "D" is plausible since the bearing is in the lube oil flowpath.

Technical Reference(s): OP3353.MB4B (Rev. 004-09) 4-2A
(Attach if not previously provided) P&ID 102A (Rev. 28)
(including version/revision number)

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: Bank #70675

Question History: Millstone 3 2002 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.8 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 5	Tier #	1	1
Loss of Reactor Coolant Makeup:	Group #	1	1
Operational implications of thermal shock to RCP seals	K/A #	APE.022.K1.01	
	Importance Rating	2.8	3.2

Proposed Question:

With the plant at 100% power with all equipment in a normal lineup, a temporary electrical disturbance occurs, and the following sequence of events occurs:

1. The "A" Charging Pump trips.
2. The "A" RPCCW Pump trips.
3. The crew enters EOP 3506 *Loss of All Charging Pumps*.
4. Simultaneously, the following two events occur
   - The STA reports affected RCP #1 Seal Inlet Temperatures have increased to 230°F.
   - The RO reports the "B" charging pump fails to start.

Which event requires the crew to immediately trip the reactor, and why?

A. The "B" Charging Pump failing to start, since there is no method available to add inventory to the RCS.

B. The "B" Charging Pump failing to start, since there is no method available to add boron to the RCS.

C. The affected RCP #1 Seal Inlet Temperature reaching 230°F, since hot seal return water creates the potential for the release of radioactive steam to the auxiliary building.

D. The affected RCP #1 Seal Inlet Temperature reaching 230°F, since the potential exists for degradation of the RCP seals, resulting in a significant increase in RCS leakage.

Proposed Answer: D

Explanation (Optional): "D" is correct, since the reactor must be tripped if all thermal barrier cooling (RPCCW or Seal Injection) is lost AND affected RCP #1 Seal Inlet Temperatures have increased to 230°F. "A" and "B" are wrong, since the reactor is not required to be tripped on a loss of all charging until PZR level drops to 9%. "A" and "B" are plausible, since inventory addition and boration capability are lost with no charging pumps running. "C" is wrong since the major concern for a loss of seal injection and thermal barrier cooling is degraded seal performance. "C" is plausible since a trip criterion is met, and the rad release concern is the basis for isolating the seal return line on a loss of all AC power.

Technical Reference(s): EOP 3506 (Rev. 009), Foldout Page
(Attach if not previously provided) EOP 3506 (Rev. 009), steps 6, 7, and 15
(including version/revision number) ERG ECA-0.0 BKGD (Rev. 2) pg 3

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06658 Discuss the basis of major precautions, procedure steps, and/or sequence of steps within EOP 3506. (As available)

Question Source: Bank #75603

Question History: Millstone 3 2001 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3, 41.8, 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 6	Tier #	1	1
Loss of RHR System:	Group #	1	1
Determine/interpret location and isolability of leaks	K/A #	APE.025.A2.04	
	Importance Rating	3.3	3.6

Proposed Question:

The plant is in MODE 5, and the following initial conditions exist:

- The crew has recently completed all steps to shift RHR from Train "B" to Train "A" in single loop operation per OP 3310A *Residual Heat Removal System*.
- The "A" Train of RHR is in service in the "Plant Cooldown" Mode.
- Letdown Flow indicates 100 gpm.

The following sequence of events occurs:

- Pressurizer level starts to drop.
- The crew enters EOP 3505 *Loss of Shutdown Cooling and/or RCS Inventory*.
- The crew takes Charging Flow Control Valve 3CHS*FCV121 to Manual, and stabilizes PZR level at 45%.
- PEOs are dispatched to locate the leak.
- A PEO reports a significant leak is coming from the RHR to CHS letdown line in the Auxiliary Building.
- The crew isolates letdown by closing RHR Letdown Flow Control Valve 3CHS-HCV128 and all three Letdown Orifice Isolation Valves.

After isolating letdown, Pressurizer level remains steady at 45%.

Has the leak been isolated from the RCS? If not, can the crew isolate the leak by closing RHR Loop A to CVCS Letdown Isolation Valve 3RHS-V20?

A. The crew has successfully isolated the leak from the Reactor Coolant System.

B. The leak has NOT been isolated from the RCS. The leak can be isolated by closing 3RHS-V20.

C. The leak has NOT been isolated from the RCS. The leak can NOT be isolated by closing 3RHS-V20, since the leak is upstream of 3RHS*V20.

D. The leak has NOT been isolated from the RCS. The leak can NOT be isolated by closing 3RHS-V20, since a leak path to the letdown line also exists from the "B" RHR Loop.

Proposed Answer: B

Explanation (Optional): Since Pzr level remains stable after isolating letdown, the leak has not been isolated from the RCS, since charging flow still exists, and letdown has been isolated. If the leak were isolated, Pzr level would start increasing ("A" wrong). Since the leak is in the letdown path in the Auxiliary Building, this is downstream of 3RHS*20, which is in the ESF Building ("B" correct, "C" wrong). "D" is wrong, since in single loop cooling, the opposite train letdown path is isolated. "A" is plausible, since Pzr level is stable, not decreasing. "C" is plausible,, since a portion of letdown piping exists upstream of 3RHS*V20. "D" is plausible, since this would be true if RHR were running in two loop operation.

Technical Reference(s): OP 3310A (Rev 016-12), section 4.9
(Attach if not previously provided) EOP 3505 (Rev 010-02), step 6
(including version/revision number) P&IDs 104A (Rev. 49) and 112A (Rev. 47)

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: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3, 41.8, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 7	Tier #	1	1
Loss of Component Cooling Water:	Group #	1	1
Operate/monitor CRDM high-temperature alarm system	K/A #	APE.026.A1.04	
Proposed Question:	Importance Rating	2.7	2.8

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

1. The CRDM SHROUD TEMPERATURE HI (VP1B, 2-4) annunciator is received.
2. The BOP operator reports that the "A" and "B" CRDM cooling fans are running, and the problem appears to be with the Reactor Plant Chilled Water (CDS) supply to the "A" CRDM Shroud Cooler.
3. Per the ARP, the US directs the BOP to inform him if Shroud ΔT exceeds 43°F.

Can the BOP directly monitor Shroud ΔT, or must he calculate it based on Shroud Inlet and Outlet Temperatures; and why shouldn't the CDS System problem result in excessive Shroud ΔT?

A. The BOP operator calculates ΔT based on CRDM Shroud Inlet Temperature at VP1 and Shroud Outlet temperature on the Plant Process Computer. CRDM Shroud ΔT should remain acceptable, since CDS cools the air after it has already flowed through the CRDM Shroud, prior to its exhausting back into CTMT.

B. The BOP operator calculates ΔT based on CRDM Shroud Inlet Temperature at VP1 and Shroud Outlet temperature on the Plant Process Computer. CRDM Shroud ΔT should remain acceptable, since, even though CDS cools the air entering the CRDM Shroud area, the "B" CRDM Shroud cooler is a 100% capacity cooler.

C. The BOP operator directly monitors CRDM Shroud ΔT at VP1. CRDM Shroud ΔT should remain acceptable, since CDS cools the air after it has already flowed through the CRDM Shroud, prior to its exhausting back into CTMT.

D. The BOP operator directly monitors CRDM Shroud ΔT at VP1. CRDM Shroud ΔT should remain acceptable, since, even though CDS cools the air entering the CRDM Shroud area, the "B" CRDM Shroud cooler is a 100% capacity cooler.

Proposed Answer: A

Explanation (Optional): CRDM Shroud inlet Temperature is read at VP1, and Shroud Outlet temperature is read on the Plant Process Computer ("C" and "D" wrong). ΔT should remain acceptable, since CDS cools the air after it has flowed through the CRDM Shroud prior to its exhausting back into CTMT, in order to assist with Containment heat removal ("A" correct, and "B" wrong). CRDM cooling is provided by air flow across the CRDM housings by the 2 50% capacity CRDM cooling fans, with or without CDS. The 3 CRDM cooling fans are 50% capacity fans, so two are likely required. "C" and "D" are plausible, since VP1 contains CRDM Cooling control and indication. "B" is plausible, since the "B" CRDM Shroud cooler is still operating and being supplied by CDS.

Technical Reference(s): OP 3353.VP1B (Rev 002-04), 2-4
(Attach if not previously provided) OP 3313C (Rev 006-02), section 1.2
(including version/revision number) P&IDs 122B (Rev. 10) and 153A (Rev. 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04257 ...Describe the Containment Ventilation System flowpath and electrical alignment under the following... conditions... Control Rod Drive System Energized... (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 8	Tier #	1	1
ATWS: Operational implications of	Group #	1	1
reactor nucleonics and thermo-hydraulic behavior	K/A #	EPE.029.K1.01	
	Importance Rating	2.8	3.1

Proposed Question:

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

1. A loss of all Main Feedwater occurs.
2. The reactor fails to trip, and the crew enters FR-S.1 *Response to Nuclear Power Generation/ATWS.*
3. The RCS heats up, and the reactor shuts down.
4. RCS temperature decreases, and the reactor returns to criticality.
5. Reactor power stabilizes at approximately 5%.
6. No operator actions have been taken.

Why did reactor power LEVEL OFF <u>at 5%</u>?

A. Doppler Power Coefficient added positive reactivity due to the cooldown.

B. A heat balance is reached based on Auxiliary Feedwater system capacity.

C. Moderator Temperature Coefficient added positive reactivity due to the cooldown.

D. A heat balance is reached based on Steam relief capacity of two atmospheric relief valves.

Proposed Answer: B

Explanation (Optional): Upon loss of feedwater, RCS heats up due to heat imbalance, shutting the reactor down. AFW is assumed to be supplied to the SGs, so the RCS starts to cool down. The reactor goes recritical, and will stabilize when heat production equals heat removal, which is limited by AFW flow to about 5% power.

Technical Reference(s): Westinghouse MITCORE Text (1991) for FR-S.1, page 2-11 and Figure 2-1.2
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04945 Assuming no Operator-initiated recovery technique, ANALYZE the ATWS Event leading to Core Damage. (As available)

Question Source: Bank # 70061

Question History: Millstone 3 2000 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.8, 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 9	Tier #	1	1
SG Tube Rupture:	Group #	1	1
Ability to operate/monitor pressurizer level/pressure	K/A #	EPE.038.EA1.09	
	Importance Rating	3.2	3.3

Proposed Question:

A SG Tube Rupture is in progress, and the following sequence of events occurs:

1. The crew enters E-3 *Steam Generator Tube Rupture.*
2. The crew reaches E-3, step 16 "Depressurize RCS To Minimize Break Flow And Refill PZR."
3. The crew commences depressurizing the RCS using maximum normal PZR spray.
4. RCS pressure drops to less than the ruptured SG pressure.
5. The crew closes the normal PZ R spray valves.
6. Prior to proceeding in the procedure, the RO is directed to monitor PZR pressure and level.

Assuming the crew is maintaining the plant in accordance with E-3, and no other pipe breaks or equipment malfunctions exist, what PZR pressure and level trend will occur?

A. PZR pressure and level will increase, since Safety Injection flow has not yet been terminated.

B. PZR pressure and level will increase, since the RCS is heating up.

C. PZR pressure and level will decrease, since primary to secondary leakage will reinitiate.

D. PZR pressure and level will decrease, since operators are cooling down the RCS at maximum rate.

Proposed Answer: A

Explanation (Optional): "A" is correct, and "C" wrong, since the depressurization step has (temporarily) stopped primary to secondary leakage, and SI is still injecting, so mass in exceeds mass out of the RCS. "C" is plausible, since primary to secondary leakage will reinitiate as RCS pressure increases. "B" is wrong, since, after previously completing the RCS cooldown, the operators were directed to maintain Core Exit TCs less than the required temperature. "B" is plausible, since if the operators were not directed to prevent a heatup, decay heat would cause RCS temperature to increase. "D" is wrong, since E-3 directs the operators to conduct the cooldown and depressurization steps sequentially, rather than concurrently. "D" is plausible, since a rapid RCS cooldown has just been conducted.

Technical Reference(s): E-3 (Rev. 021), steps 6.f, 13.c, 16, and 17.
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04919 Describe the major parameter changes associated with SGTRs. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 10	Tier #	1	1
Steam Line Rupture: Determine/interpret difference between steam line break and LOCA	Group #	1	1
	K/A #	APE.040.A2.03	
	Importance Rating	4.6	4.7

Proposed Question:

With the plant initially stable at 100% power, the following annunciator is received:

TREF/AUCT TAVE DEVIATION (MB4C, 6-5).

The RO commences reporting primary plant parameters, and his first report is as follows:

PARAMETER:	CURRENT VALUE:	TREND:
Pressurizer Level:	59%	Decreasing

Based on these conditions, what event is in progress?

A. Small break RCS LOCA

B. Steamline break

C. Main Turbine Runback

D. Steam Generator Tube Rupture

Proposed Answer: B

Explanation (Optional): Based on the decrease in PZR Level, either a loss of inventory event is in progress ("A" and "D" plausible), or a heat imbalance exists ("C" plausible). "A" and "D" are wrong, since a loss of RCS inventory will not result in a Tave Deviation. "B" is correct, and "C" wrong, since Pressurizer Level is decreasing, indicating that the temperature deviation is due to a cooldown, showing an increase in heat removal (or a decrease in heat production, but none of the distractors would cause this).

Technical Reference(s): FSAR (Rev. 21.3) Figures 15.1-15 and 15.1-16
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04881 DESCRIBE the major parameter changes associated with increased heat removal by the Secondary System. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1, 41.5, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 11	Tier #	1	1
Loss of Main Feedwater: Operational implications of	Group #	1	1
effects of feed introduction into a dry steam generator	K/A #	APE.054.K1.02	
Proposed Question:	Importance Rating	3.6	4.2

With the plant initially at 100% power and the TDAFW pump tagged out, the following sequence of events occurs:

1. The "A" Feed Reg Valve fails closed.
2. The reactor trips on Lo-Lo level in the "A" SG.
3. Both MDAFW pumps start.
4. The BOP reports AFW flow can NOT be established to the "A" SG, since MDAFW flow control valve to the "A" SG (3FWA*HIC31A1) has failed closed.
5. The crew completes ES-0.1 *Reactor Trip Response* and transitions to FR-H.5 *Response to Steam Generator Low Level.*

Maintenance reports that the MDAFW flow control valve to the "A" SG has been repaired, and current conditions are as follows:

- RCS Tave: 557°F
- Total AFW flow: 600 gpm
- All four SG NR Levels: Off-scale low
- "A" SG Wide Range level: 3%

Will FR-H.5 direct the crew to establish AFW flow to the "A" SG? Why or why not?

A. The crew WILL establish AFW flow to the "A" SG. AFW flow is desired to establish even cooling to all four RCS loops.

B. The crew WILL establish AFW flow to the "A" SG. AFW flow is required to establish minimum heat sink with all SG levels below 8% narrow range.

C. The crew will NOT establish AFW flow to the "A" SG. Increasing AFW flow will cool down the plant, creating a Shutdown Margin concern.

D. The crew will NOT establish AFW flow to the "A" SG. Adding AFW to the "A" SG will create significant thermal stresses on SG components.

Proposed Answer: D

Explanation (Optional): Feeding a hot (>550°F), dry (WR level <12%) will create significant thermal stresses on SG components. Therefore, FR-H.5 directs the crew to request the ADTS to evaluate refilling the affected SG as part of long term recovery actions, and transitions out of FR-H.5, skipping the step to restore AFW flow ("D" correct, "A", wrong). "A" is plausible, since even cooling is desired for normal cooldowns. "B" is wrong, since total AFW flow is adequate for heat sink, but plausible, since all SG NR levels are offscale low. "C" is wrong, since the other SGs can be throttled back if feed is introduced to the "A" SG, but plausible, since cooling down adds positive reactivity.

Technical Reference(s): FR-H.5 (Rev. 008), steps 4 and 5
(Attach if not previously provided) BOG Bkgd Document (Rev 2) for FR-H.5, step 4
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05975 Discuss the basis of major procedure steps and/or sequence of steps associated with EOP FR-H.5 (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.8 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 12	Tier #	1	1
Station Blackout: Ability to operate/monitor	Group #	1	1
the reduction of loads on the batteries	K/A #	EPE.055.A1.04	
	Importance Rating	3.5	3.9

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

1. A loss of all AC Power occurs.
2. The crew enters ECA-0.0 *Loss of All AC Power*.
3. The crew reaches ECA-0.0, step 16 "Check DC Bus Loads."
4. A PEO is dispatched to carry out the local actions in step 16.

Assuming all equipment operates as expected, what actions will the PEO take, and what indications will the BOP operator see in response to the PEO's actions?

A. The PEO will open the main breakers supplying power to non-essential DC Panels, and the BOP operator will see an associated decrease in battery discharge currents for Batteries 1, 2, 3, and 4 on MB8.

B. The PEO will open the main breakers supplying power to non-essential DC Panels, and the BOP operator will see an associated decrease in battery discharge currents for Batteries 1, 2, 5, and 6 on MB8.

C. The PEO will remove the individual fuse blocks for each of the loads on the non-essential DC Panels, and the BOP operator will see an associated decrease in battery discharge currents for Batteries 1, 2, 3, and 4 on MB8.

D. The PEO will remove the individual fuse blocks for each of the loads on the non-essential DC Panels, and the BOP operator will see an associated decrease in battery discharge currents for Batteries 1, 2, 5, and 6 on MB8.

Proposed Answer: B
Explanation (Optional): The PEO will open the main breakers to the DC panels ("C" and "D" wrong), decreasing output current. "B" is correct, and "A" wrong, since load is removed from Batteries 1, 2, 5, and 6, since Battery busses supply only their associated inverters. "A" is plausible, since batteries 1, 2, 3, and 4 supply vital busses. "C" and "D" are plausible; since the Response Not Obtained column has the operators remove the individual fuse blocks for loads on the non-essential DC Panels if the crew is unable to open the main breaker(s).

Technical Reference(s): ECA-0.0 (Rev. 021), step 16.a.
(Attach if not previously provided) ECA-0.0 (Rev. 021), Attachment A, page 1 of 19.
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03851 Describe the major action categories within EOP 35 ECA-0.0. (As available)
MG-00724 Solve for various parameters in various AC and DC circuits…

Question Source: New
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 13	Tier #	1	1
Loss of Vital AC Elec. Inst. Bus:	Group #	1	1
Determine/interpret valve indicator of charging pump suction valve from RWST	K/A #	APE.057.AA2.07	
	Importance Rating	3.3	3.5

Proposed Question:

With the plant initially at 100% power with PZR level control selected to Channel I-II, the following sequence of events occurs:

1. VIAC 1 deenergizes.
2. The operators take initial actions to take manual control and stabilize the plant.
3. The crew enters AOP 3564 *Loss of One Protective System Channel*.
4. The RO reports that VCT level indicates 40% (using computer point CHS—L112).
5. The crew determines that AUTO Makeup has not initiated, due to a loss of power to the Reactor Coolant Makeup Control Auxiliary Circuit.

Assuming no further operator actions (other than taking manual control of controllers) have been taken, how will VCT level respond to the loss of VIAC 1?

A. VCT level will remain stable, since Charging plus Seal Injection flows are matched with Letdown and Seal Leakoff flows.

B. VCT level will remain stable, since RWST to Charging Pump Suction Valves 3CHS*LCV 112D and 112E automatically OPENED when VIAC 1 deenergized.

C. VCT level will continue to drop at a fairly rapid rate, since Letdown isolated when VIAC 1 deenergized. At 4% VCT level, RWST to Charging Pump Suction Valves 3CHS*LCV 112D and 112E will automatically OPEN.

D. VCT level will continue to drop at a fairly rapid rate, since Letdown isolated when VIAC 1 deenergized. At 4% VCT level, RWST to Charging Pump Suction Valves 3CHS*LCV 112D and 112E will NOT automatically OPEN.

Proposed Answer: C

Explanation (Optional): On a loss of VIAC 1, Pressurizer Level Channel 459 fails low, causing letdown to isolate (1 of 2 channels) ("A" wrong). This results in a decreasing VCT level, since charging and seal injection are still in service. VCT level transmitters LT112 and 185 do not lose power on a loss of VIAC 1 ("B" is wrong), so, at 4% VCT level (2 of 2 channels) Charging Pump Suction Valves 3CHS*LCV 112D and 112E will automatically OPEN, and VCT outlet valves 3CHS*LCV112B and C will automatically CLOSE, to maintain Charging Pump suction ("C" correct, "D" wrong). "A" is plausible, since this would be true if VIAC one did not supply the PZR level channel that isolates letdown, or if the coincidence was 2 of 2 channels. "B" is plausible, since this would be true if VIAC one supplied power to the VCT level transmitters. "D" is plausible, since this would be true if the loss of VIAC 1 affects PZR level indication, and the makeup system.

Technical Reference(s):	AOP 3564 (Rev. 009-02), step 5.
(Attach if not previously provided)	ESK-7CR (Rev. 13)
(including version/revision number)	LSK 25-1.2D (Rev. 7), 26-2.1A (Rev. 8), 26-2.2C (Rev. 8) and 26-2.2D (Rev. 8)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07021 For the below listed failures, partial or complete, describe the effects on the Primary Makeup System… Loss of Power… (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 14	Tier #	1	1
Loss of Service Water: Reasons for automatic alignments of service water resulting from ESFAS actuation	Group #	1	1
	K/A #	APE.062.AK3.02	
	Importance Rating	3.6	3.9

Proposed Question:

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

1. An inadvertent CDA occurs.
2. The crew enters E-0 *Reactor Trip or Safety Injection*.
3. While walking down his boards, the RO observes that Service Water has been lost to the RPCCW Heat Exchangers, since the Service Water Supply Valves to RPCCW (3SWP*MOV50A and B) automatically closed.

Why were 3SWP*MOV50A and B designed to automatically close on the CDA Signal?

A. This prevents excessive flow conditions in the Service Water System while supplying RSS.

B. This prevents robbing flow from the EDG Service Water Coolers in the event of an LOP.

C. This allows adequate pressure to refill the Control Building Chiller Service Water Booster Pump suction piping.

D. This allows adequate pressure to refill the MCC/Rod Control Area Service Water Booster Pump suction piping.

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since the flow required to supply both the RSS System and the RPCCW system is beyond the capacity of a Service Water Pump. Service Water Pumps should not be operated above 15,000 gpm, and flow to an RPCCW Heat Exchanger is about 8000 gpm, and flow to a train of RSS Heat Exchangers is about 10,000 gpm. "B" is plausible, since the EDG cooling water valves automatically open on a CDA, to provide cooling to the EDGs, which automatically started. "C" and "D" are plausible, since both the Control Building Chiller Booster Pumps and the MCC/Rod Control Booster Pumps are at a high elevation. Booster Pump priming is accomplished by the time delay associated with the automatic opening of Service Water to RSS Heat Exchangers C and D (3SWP*MOV 54C and D).

Technical Reference(s): OP 3326 (Rev. 023-02), Precaution 3.8
(Attach if not previously provided) FSAR (Rev. 21.3), Table 9.2-1
(including version/revision number) Millstone 3 Training Lesson Plan SWP076C (Rev. 3, Ch. 3), Pages 23-26

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05714 Describe the operation of the following Service Water System components, controls, and interlocks… RPCCW Heat Exchanger Isolation Valves… (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4 and 41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 15	Tier #	1	1
Loss of Instrument Air: Determine/interpret	Group #	1	1
failure modes of air-operated equipment	K/A #	APE.065.AA2.08	
	Importance Rating	2.9	3.3

Proposed Question:

The plant is initially in MODE 5 with the following conditions present:

- The Pressurizer is solid.
- The "A" RHR train is in service.
- RCS Temperature is stable at 145°F.

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

Assuming NO operator action, what effect will the loss of instrument air pressure have on RCS temperature, and why?

A. RCS temperature will increase due to decreased RPCCW flow through the RHR Heat Exchanger.

B. RCS temperature will increase due to decreased RHR flow through the RHR Heat Exchanger.

C. RCS temperature will decrease due to increased RPCCW flow through the RHR Heat Exchanger.

D. RCS temperature will decrease due to increased RHR flow through the RHR Heat Exchanger.

Proposed Answer: D

Explanation (Optional): 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. 006), Page 3
(Attach if not previously provided)
(including version/revision number)

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: Bank #73098

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.7 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 16	Tier #	1	1
LOCA Outside Containment: Operate/monitor instruments,	Group #	1	1
signals, interlocks, failure modes, auto/manual features	K/A #	EPE.W/E04.A1.01	
Proposed Question:	Importance Rating	4.0	4.0

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

1. A LOCA outside Containment occurs, resulting in a reactor trip and safety injection.
2. Over the next 10 minutes, RCS pressure increases to 2350 psia, and the PZR PORVs start cycling.
3. The crew is responding using ECA-1.2 *LOCA Outside Containment.*
4. While attempting to isolate the break, the final valve the crew is preparing to close is SI Injection Valve 3SIH*MV8835.
5. Just prior to closing 3SIH*MV8835, the RO reports the following conditions:
   - Pzr level is 65% and increasing.
   - Pzr pressure is cycling at 2350 psia.
6. After 3SIH*MV8835 closes, the RO reports that the PORVs are cycling at a significantly faster rate based on Real-Time indication.

What is the status of the leak, and what indication can the RO monitor to provide a reliable, diverse indication of leak status?

A. The LOCA outside CTMT is still active. A reliable, diverse indication that the leak is still active is the Pzr level trend, which should be increasing at the same rate as before.

B. The LOCA outside CTMT is still active. A reliable, diverse indication that the leak is still active is the RCS pressure trend, which should be trending down toward 2250 psia.

C. The LOCA outside CTMT has been isolated. A reliable, diverse indication that the leak is isolated is the Pzr level trend, which should be increasing at a faster rate.

D. The LOCA outside CTMT has been isolated. A reliable, diverse indication that the leak is isolated is the RCS pressure trend, which should be increasing.

Proposed Answer: C

Explanation (Optional): The NOTE prior to step 1 must be applied while pressure is cycling on the PORVs. This is required since, for smaller breaks, ECCS flow may cause an RCS pressure increase with or without break isolation, and the procedurally directed use of a pressure increase to determine leak status may not be work, since the RCS is cycling on the PORVs ("B" and "D" wrong). Other means of verifying break isolation should be checked such as pressurizer level increase ("C" correct), reports from the field, decrease in area radiation, or an increase in PORV cycling frequency ("A" and "B" wrong). Pzr pressure will remain at 2350 psia, whether or not the break is isolated, since letdown is isolated, and seal injection is still entering the RCS ("A" and "B" plausible). "D" is plausible, since RCS pressure increasing is the normal way the leak status is checked per the step.

Technical Reference(s):	ECA-1.2 (Rev. 8), Note prior to step 1
(Attach if not previously provided)	ECA-1.2 (Rev. 8), steps 4 and 5
(including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07434 Given a set of plant conditions, properly apply the notes and cautions of ECA-1.2. (As available)

Question Source: New

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 # 17	Tier #	1	1
Loss of Emergency Coolant Recirc:	Group #	1	1
Interrelations between instruments, signals, interlocks,	K/A #	EPE.W/E11.K2.01	
failure modes, and auto/manual features	Importance Rating	3.6	3.9

Proposed Question:
With the plant initially at 100% power, a large break LOCA occurs, and the following sequence of events occurs:

- The crew is unable to establish emergency coolant recirculation.
- The crew enters ECA-1.1 *Loss of Emergency Coolant Recirculation*.
- The US directs the RO to complete ECA-1.1, Attachment B "Establish Cold Leg Recirculation."
- While performing the lineup the RO places the switch for RHR to CHG and SI Suction Isolation Valve 3SIL*MV8804A to OPEN, but the valve remains closed.

Which abnormal valve position could have prevented RHR to CHG and SI Suction Isolation Valve 3SIL*MV8804A from opening?

A. Charging Pump Miniflow isolation valves to the RWST 3CHS*MV8511A and B are CLOSED.

B. RSS to RHR Isolation Valves 3RSS*MV8837A and 8838A are CLOSED.

C. RHR Loop Suction Isolation Valves 3RHS*MV8701A, 8701B, and 8701C are CLOSED.

D. SI Pump Miniflow Isolations 3SIH*MV8813, 8814 AND MV8920 are CLOSED.

Proposed Answer: B

Explanation (Optional): The interlocks required to OPEN 3SIL*MV8804A are: SI Pump Miniflow Isolations 3SIH*MV8813 CLOSED, OR 8814 AND MV8920 CLOSED ("D" wrong); Charging Pump Miniflow isolation valves 3CHS*MV8511A and B OR 8512A and B CLOSED ("A" wrong), RHR Loop Suction Isolation Valves 3RHS*MV8701A, 8701B, OR 8701C CLOSED ("C" wrong), and RSS to RHR Isolation Valves 3RSS*MV8837A or 8838A OPEN ("B" correct). "A", "C", and "D" are plausible, since they are all part of the SIL*MV8804A interlock.

Technical Reference(s): ECA-1.1 (Rev. 016) Attachment B, step 2
(Attach if not previously provided) LSK 27-3C (Rev. 14)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05455 Describe the operation of the following RHR System Controls and Interlocks… RHR to CHS/SIH Pump Supply Valves (3SIL*MV8804A/B)… (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 18	Tier #	1	1
Generator Voltage and Electric Grid Disturbances:	Group #	1	1
Operational implications of over-excitation	K/A #	APE.077.AK1.02	
	Importance Rating	3.3	3.4

Proposed Question:

With the plant initially at 55% power, the following annunciator is received:

- GENERATOR OVER EXCITATION (MB7C, 5-5)

In accordance with the associated ARP, what operational implications exist?

A. The BOP operator must repeatedly toggle the "VOLT REG" switch (MB7) to the LOWER position to clear the alarm; and if unsuccessful, the crew is required to trip the reactor and go to E-0 *Reactor Trip or Safety Injection*.

B. The BOP operator must repeatedly toggle the "VOLT REG" switch (MB7) to the LOWER position to clear the alarm; and if unsuccessful, the crew is required to trip the main turbine and go to AOP 3550 *Turbine/Generator Trip*.

C. The BOP operator must depress and hold the "DECREASE LOAD" pushbutton (MB7, EHC Insert) to the LOWER position to clear the alarm; and if unsuccessful, the crew is required to trip the reactor and go to E-0 *Reactor Trip or Safety Injection*.

D. The BOP operator must depress and hold the "DECREASE LOAD" pushbutton (MB7, EHC Insert) to the LOWER position to clear the alarm; and if unsuccessful, the crew is required to trip the main turbine and go to AOP 3550 *Turbine/Generator Trip*.

Proposed Answer: A

Explanation (Optional): "A is correct, since the BOP operator is directed to reduce load using the "VOLT REG" "MAN/AUTO" switch, and if unsuccessful with power above P-9 (51% power), the crew is required to trip the reactor and go to E-0 *Reactor Trip or Safety Injection*. "B" is wrong, but plausible, since this would be correct if power were less than P-9. "C" and "D" are wrong, but plausible, since this action will lower real load, rather than excitation.

Technical Reference(s): OP 3353.MB7C (Rev. 003-05), Section 5-5
(Attach if not previously provided) Shift Brief SB3-04-014 (Rev. 0)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04682 Describe the operation of the Main Generator, Exciter and Regulator components controls and interlocks… (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.5, 41.7, and 41.10

Comments:

Examination Outline Cross-reference:		Level	RO	SRO
Question # 19		Tier #	1	1
Dropped Control Rod: Operate/monitor		Group #	2	2
the demand counter and the P/A converter		K/A #	APE.003.A1.01	
Proposed Question:		Importance Rating	2.9	2.9

While operating at 100% power, the following sequence of events occurs:

1. One Control Bank D, Group 1 rod drops fully into the core.
2. The crew enters AOP 3552 *Malfunction of the Rod Drive System.*
3. I&C corrects the cause of the dropped rod.
4. The crew is preparing to recover the rod

With regard to rod position indication, what actions are required to be taken by the crew with the group step counters, and with the P/A converter?

A. Prior to aligning the rod, they will reset the affected group step counter to zero, and use the group counter to determine how far out to withdraw the rod. After aligning the rod, they will dispatch a PEO to restore the P/A converter by adjusting its display to agree with the affected group 1 counter demand position.

B. Prior to aligning the rod, they will leave the affected group step counter at its current height, since the bank will not be moving as the rod is withdrawn. After aligning the rod, they will dispatch a PEO to restore the P/A converter by adjusting its display to agree with the affected group 1 counter demand position.

C. Prior to aligning the rod, they will reset the affected group step counter to zero, and use the group counter to determine how far out to withdraw the rod. After aligning the rod, they will not adjust the P/A converter, since it monitors Group 2 rod demand signals, and Group 2 did not move during the recovery.

D. Prior to aligning the rod, they will leave the affected group step counter at its current height, since the bank will not be moving as the rod is withdrawn. After aligning the rod, they will not adjust the P/A converter, since it monitors Group 2 rod demand signals, and Group 2 did not move during the recovery.

Proposed Answer: A

Explanation (Optional): The crew resets the affected group counter to zero, since that is where the affected rod is located after dropping, and the rod will be withdrawn to match the original group height of the bank ("B" and "D" wrong). The P/A converter monitors the demand signal for the group 1 control bank rods only, and transfers those signals to the RIL circuitry as the total bank demand signals. Since the rod is a group 1 rod, the crew is required to restore the P/A converter, since, it will be counting as the group one rod is withdrawn, causing the P/A converter to get out of step with the bank ("A" is correct, and "C" wrong). "B" and "D" are plausible, since it is true for affected bank DRPI. "C" is plausible, since this would be true if the rod was a group 2 rod, or a shutdown bank rod.

Technical Reference(s): AOP 3552 (Rev. 009), Attachment B, steps 4.c and 6.g.
(Attach if not previously provided) AOP 3552 (Rev. 009), Attachment E.
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05484 Describe the operation of the Rod Position Indication System under the following Normal, Abnormal, and Emergency conditions… Stuck, Misaligned, or Dropped Rod (including recovery operations)… (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 # 20	Tier #	1	1
Loss of Source Range NI: Operational implications of	Group #	2	2
the effect of voltage changes on performance	K/A #	APE.032.K1.01	
Proposed Question:	Importance Rating	2.5	3.1

Initial conditions:

- A reactor startup is in progress in accordance with OP 3202 *Reactor Startup*.
- Control banks are being withdrawn
- Source Range counts: N31 = 1500 cps
N32 = 1600 cps

A faulty power supply causes significant instrument power voltage fluctuations to Source Range channel N31.

Which of the following describes the effect of the power supply voltage fluctuations on the plant?

A. Instrument voltage changes will result in proportional changes in indicated count rate. The reactor may trip on high source range flux.

B. Instrument voltage changes will result in proportional changes in indicated count rate. The reactor will not trip since the source range high flux trip is already blocked.

C. Instrument voltage changes will not result in changes in indicated count rate, since the detector operates in the ion chamber region. The crew will continue the startup.

D. Instrument voltage changes will not result in changes in indicated count rate, since the detector operates in the ion chamber region. The crew will stop the startup and drive rods in.

Proposed Answer: A

Explanation (Optional): "C" and "D" are wrong since the source ranges operate in the proportional region of the gas amplification curve. The Source Range high flux trip will actuate when counts exceed 10^5 cps on 1/2 channels. Since the coincidence is 1/2, the reactor will trip. "A" is correct, and "B" is wrong, since source ranges are blocked above P-6, which comes in at 10^{-10} amps in the intermediate range, which is above 10^4 cps. "B" is plausible since source ranges will be blocked during the startup. "C" and "D" are plausible since the intermediate range detectors operate in the ion chamber region.

Technical Reference(s): OP 3360 (Rev. 007-06), Precautions 3.3 and 3.4
(Attach if not previously provided) OP 3202 (Rev. 021), step 4.28
(including version/revision number) NIS015C (Rev. 3 Ch 2), pg 6 and 7

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05229 For the following conditions, determine the effects on the NIS system and on interrelated systems: Source range instrument failure below P-6... (As available)

Question Source: Bank # 75606

Question History: Millstone 3 2001 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.6, 41.7, 41.8

Comments:

Examination Outline Cross-reference:		Level	RO	SRO
Question # 21		Tier #	1	1
Loss of Intermediate Range NI: Operate/monitor power-available indicators in cabinets or equipment drawers		Group #	2	2
		K/A #	APE.033.A1.01	
		Importance Rating	2.9	3.1

Proposed Question:

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

1. An IR LOSS OF DET VOLTAGE annunciator is received on MB4C.
2. The RO goes over to the Intermediate Range NIS drawer to investigate.

What indications will the RO observe on the affected IRNI drawer?

A. The CONTROL POWER ON light will be dark. The HIGH LEVEL TRIP bistable light will be dark on the affected IRNI drawer.

B. The CONTROL POWER ON light will be dark. The HIGH LEVEL TRIP bistable light will be lit on the affected IRNI drawer.

C. The INSTRUMENT POWER ON light will be dark. The HIGH LEVEL TRIP bistable light will be dark on the affected IRNI drawer.

D. The INSTRUMENT POWER ON light will be dark. The HIGH LEVEL TRIP bistable light will be lit on the affected IRNI drawer.

Proposed Answer: D

Explanation (Optional): An instrument power fuse has blown, since instrument power supplies detector high voltage ("A" and "B" wrong). The bistable light will be lit, since instrument power supplies the high level trip bistable, which will fail to the trip condition, and control power supplies the bistable lights, which will still function ("D" correct, "C" wrong). "A" and "B" are plausible, since the control power supplies a portion of IRNI circuit. "C" is plausible, since the bistable lights would be dark if control power were lost.

Technical Reference(s):	OP 3353.MB4C (Rev. 006-00), 4-2
(Attach if not previously provided)	NIS Tech Manual Drawing for Control Power NIS015T-02 (Rev. 0)
(including version/revision number)	Functional Sheet 3 (Rev. G)
	Intermediate Range Drawer training drawing NIS015T-020 (Rev. 1)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05229 For the following conditions, determine the effects on the Nuclear Instrumentation System and on interrelated systems… Intermediate Range instrument failure above P-10… (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 22	Tier #	1	1
Fuel Handling Accident: Knowledge of the reasons for guidance contained in the EOP for fuel handling incident	Group #	2	2
	K/A #	APE.036.AK3.03	
	Importance Rating	3.7	4.1

Proposed Question:

Current Conditions:

- A fuel handling accident has occurred in the Fuel Building.
- The crew has entered EOP 3502 *Fuel Handling Accident*.
- Per EOP 3502, step 11, the crew is verifying that a Fuel Building Filter Unit (3HVR*FN10A or B) is running.

For what reason does EOP 3502 direct the crew to make this check?

A. Remove at least 99% of the iodine gap activity released from the ruptured fuel assembly.

B. Limit the maximum gamma dose rate in the fuel building to 2.5 mrem per hour.

C. Prevent the high levels of radioactivity from entering the Control Room.

D. Minimize the potential radioactive release to the environment.

Proposed Answer: D

Explanation (Optional): "D" is correct, since the Fuel Building Filters contain charcoal which will remove iodine from the fuel building air prior to exhausting it to the environment. "A" is wrong, but plausible, since this is the reason 23 feet of water is maintained over the fuel in the spent fuel pool. "B" is wrong, but plausible, since this is the reason at least 10.5 feet of water is maintained over spent fuel that is being moved. "C" is wrong, but plausible, since this is the reason operators will actuate CBI per EOP 3502.

Technical Reference(s): EOP 3502, step 11
(Attach if not previously provided) FSAR (Rev. 21-3) Section 9.4.2
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06415 Describe the operation of the following Fuel Handling System… Interlocks… (As available)

Question Source: New

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 # 23	Tier #	1	1
Steam Generator Tube Leak: Determine/interpret	Group #	2	2
status of Tube Leak, using independent, redundant	K/A #	APE.037.A2.09	
condensate air ejector exhaust monitor	Importance Rating	2.8	3.4

Proposed Question:

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

1. N-16 Radiation Monitor, MSS-RE80A, goes into ALERT status and the RO reports its trend history is NOT normal.
2. The crew enters AOP 3576 *Steam Generator Tube Leak.*
3. Chemistry is dispatched to sample all 4 SGs for activity.
4. Air Ejector Rad Monitor ARC21-1 starts trending upward, and the crew anticipates that it will exceed the ALERT setpoint in about 5 minutes.
5. Chemistry reports their initial sample results should be available in about 5 minutes.
6. The crew is currently is at AOP 3576, step 3 "Verify Primary to Secondary Leakage."

How do the above conditions affect progress through AOP 3576?

A. Leakage <u>is</u> verified, based on two indications of primary to secondary leakage. Continue on to Step 4.

B. Leakage is <u>NOT</u> verified. Wait for Chemistry's sample to confirm the presence of primary to secondary leakage.

C. Leakage is <u>NOT</u> verified. Wait to see if ARC21-1 exceeds the ALERT setpoint to confirm the presence of primary to secondary leakage.

D. Leakage is <u>NOT</u> verified. Request Chemistry to commence SP3861 *Primary to Secondary Leak Rate Determination* and exit AOP 3576.

Proposed Answer: A

Explanation (Optional): AOP 3576 Note prior to step 3 requires 2 indications. ARC21 Rad Trend NOT Normal AND MSS80 in Alert satisfy the requirement ("A" correct) even without the chemistry sample results. Since 2 criteria are satisfied, no further indications are required ("B" and "C" wrong). "D" is wrong, but plausible, since this action is not required unless it is determined that a plant shutdown is not required per AOP 3576, steps 7 and 8. "B" and "C" are plausible, since ARC21 has not yet reached the ALERT setpoint.

Technical Reference(s): Reference: AOP 3576 (Rev. 003), steps 3, 7, and 8.
(Attach if not previously provided) OP 3272 (Rev. 008-08), Attachment 5 Definition of Normal Radiation.
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07572 Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of AOP 3576. (As available)

Question Source: Bank #72474

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 # 24	Tier #	1	1
Inadequate Core Cooling: Operational implications of annunciators, conditions, indications and remedial actions associated with saturated core cooling conditions	Group #	2	2
	K/A #	EPE.W/E07.K1.03	
	Importance Rating	3.2	3.6

Proposed Question:

An earthquake occurs, resulting in the following:

- A small break LOCA occurs.
- A tube rupture occurs in the "D" SG

The following sequence of events occurs:

1. The crew enters ECA-3.2 *SGTR with Loss of Reactor Coolant - Saturated Recovery Desired.*
2. While progressing through ECA-3.2, a Yellow Path comes in on the Core Cooling status tree, flagging FR-C.3 *Response to Saturated Core Conditions.*
3. The STA reviews FR-C.3, and reads the following NOTE:

   "DO NOT use this procedure, if ECA-3.2, SGTR With Loss of Reactor Coolant – Saturated Recovery Desired, is in progress."

In accordance with the WOG Background for FR-C.3, why should the crew NOT implement FR-C.3 during this event?

A. FR-C.3 directs re-establishing ECCS flow to increase subcooling. This conflicts with the efforts of ECA-3.2.

B. FR-C.3 does not direct a transfer to ES-1.3 *Transfer to Cold Leg Recirculation* if RWST Lo-Lo level occurs, and a LOCA is in progress.

C. FR-C.3 directs isolation of both letdown and excess letdown. This conflicts with the efforts of ECA-3.2.

D. FR-C.3 does not limit RCS cooldown rate to less than 80°F per hour, and could lead to a Pressurized Thermal Shock condition.

Proposed Answer: A

Explanation (Optional): This question is considered an RO level question, since it deals with big-picture FR-C.3 strategy, rather than deep assessment of current plant conditions. "A" is correct, since FR-C.3 will attempt to restore ECCS flow to recover from saturated core conditions, and this conflicts with ECA-3.2, which intentionally reduces ECCS flow and subcooling, to reduce pressure and minimize break flow. "B" is wrong, since FR-C.3 does direct a transition to ES-1.3, if required, and plausible, since, ES-1.3 is a PRA significant procedure. "C" is wrong, since letdown is not the concern with using FR-C.3 with ECA-3.2. "C" is plausible, since isolating letdown is directed in FR-C.3. "D" is wrong, since FR-C.3 does not direct a change in cooldown rate, but plausible, since ECA-3.2 has directed an 80°F/hr rate, which prevents PTS.

Technical Reference(s): FR-C.3 (Rev. 006), step 1 NOTE.
(Attach if not previously provided) WOG Background Document (Rev. 2), FR-C.3 note prior to step 1.
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05975 Discuss the basis of major procedure steps and/or sequence of steps in EOP FR-C.3. (As available)

Question Source: Bank #70465

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.8, 41.10, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 25	Tier #	1	1
Rediagnosis: Determine/interpret adherence	Group #	2	2
to appropriate procedures and operation within license	K/A #	EPE.W/E01.EA2.2	
	Importance Rating	3.3	3.9

Proposed Question:

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

1. The reactor trips.
2. The crew enters E-0 *Reactor Trip or Safety Injection*.
3. The crew transitions to ES-0.1 *Reactor Trip Response*.
4. With RCS pressure and temperature stable, Air Ejector Radiation Monitor 3ARC-21 goes into ALARM.

Is the crew required/allowed to transition to ES-0.0 *Rediagnosis* to address the potential tube leak?

A. No. The crew is required to remain in ES-0.1. The crew is NOT allowed to enter ES-0.0 since they have already exited E-0.

B. No. The crew is required to remain in ES-0.1. The crew is NOT allowed to enter ES-0.0 unless SIS has actuated.

C. Yes. The crew is allowed to enter ES-0.0 based on operator judgment, anytime after entering the EOP network.

D. Yes. The crew is required to enter ES-0.0 based on current symptoms meeting its specific entry conditions.

Proposed Answer: B

Explanation (Optional): This question is considered an RO level question, since it deals with big-picture ES-0.0 rules of usage, rather than deep assessment of current plant conditions. ES-0.0, "Rediagnosis" has no symptoms or entry conditions and is entered solely based on operator judgment ("D" wrong) if SI is actuated. This is a unique procedure which may be used as an aid to determine the correct path through the EOP network after exiting E-0 ("A" wrong) with SI actuated ("B" correct and "C" wrong).

Technical Reference(s): OP 3272 (Rev. 008-08), last paragraph in section 1.2
(Attach if not previously provided)
(including version/revision number)
Proposed references to be provided to applicants during examination: None
Learning Objective: MC-04451 Discuss the conditions under which ES-0.0 can be used. (As available)
Question Source: New
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 26	Tier #	1	1
Steam Generator Over-pressure:	Group #	2	2
Ability to determine operability and/or availability of	K/A #	EPE.W/E13.GEN.2.2.37	
safety related equipment	Importance Rating	3.6	4.6

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

1. A spurious MSI actuates (and cannot be reset), and the reactor trips.
2. The crew enters ES-0.1 *Reactor Trip Response*.
3. The crew enters FR-H.2 *Response To Steam Generator Overpressure* due to "A" SG pressure exceeding 1220 psig.

The crew is currently attempting to dump steam from the "A" SG via the following paths per FR-H.2, step 4:

- The "A" Atmospheric Relief Valve
- The "A" Atmospheric Relief Bypass Valve
- The "A" MSIV Bypass Valve
- The "A" Steam Supply to the TDAFW pump

Which of these valves should either be already open, or available to be opened by the operators in the control room?

A. The Atmospheric Relief Valve should already be open. The Atmospheric Relief Bypass Valve is available to be opened.

B. The Atmospheric Relief Valve should already be open. The MSIV Bypass Valve is available to be opened.

C. The Steam Supply to the TDAFW pump should already be open. The Atmospheric Relief Bypass Valve is available to be opened.

D. The Steam Supply to the TDAFW pump should already be open. The MSIV Bypass Valve is available to be opened.

Proposed Answer: C

Explanation (Optional): An MSI signal automatically closes the MSIVs, the MSIV bypass valves ("B" and "D" wrong), and the atmospheric relief valves ("A" wrong). "C" is correct, since the atmospheric relief bypass valve is not impacted by an MSI, and the steam supply valves to the TDAFW Pump automatically open on the SG lo-lo levels resulting from shrink on the reactor trip, and do not receive an MSI auto-close signal. "A" is plausible, since the steam pressure is above the auto-open setpoint (1100 psig) of the atmospheric relief valves. "B" and "D" are plausible, since the MSIV bypass valves are a normally-closed bypass around the MSIVs.

Technical Reference(s): FR-H.2 (Rev 009), step 4
(Attach if not previously provided) P&ID 123A (Rev. 48), B (Rev. 24), and D (Rev. 14)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05965 Describe the major action categories within EOP 35 FR-H.2 (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 27	Tier #	1	1
Turbine Trip: Knowledge of alarms,	Group #	2	2
indications, or annunciator response procedures	K/A #	Site Specific: Turbine Trip.GEN.2.4.31	
	Importance Rating	4.2	4.1

Proposed Question:

With the plant initially at 40% power, the following sequence of events occurs:

1. The Main Turbine trips.
2. The crew enters AOP 3550 *Turbine/Generator Trip*.
3. The US directs the BOP operator to "Check if Condenser Vacuum Should Be Broken."

In accordance with AOP 3550, which of the following Turbine First-Out annunciators (MB7B) would require the crew to open the condenser vacuum breakers if the parameter cannot be restored?

A. LOSS OF STATOR COOLANT

B. EXH HOOD TEMP HI-HI

C. MOIST SEP WTR LEVEL HI

D. BEARING OIL PRES LOW

Proposed Answer: D

Explanation (Optional): Condenser vacuum will be broken for any of the following: Hi turbine vibration, thrust bearing failure, bearing oil low pressure ("D" correct), gland steam pressure low, or indications of turbine failure ("A", "B", and "C" wrong). "A", "B", and "C" are plausible, since they are all first out annunciators that cause an automatic turbine trip.

Technical Reference(s): AOP 3550 (Rev. 007-04), step 8
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03896 Describe the major action categories within AOP 3550. (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 28	Tier #	2	2
Reactor Coolant Pump:	Group #	1	1
Effect of a loss or malfunction of RCPs on S/Gs	K/A #	003.K3.02	
	Importance Rating	3.5	3.8

Proposed Question:

With the plant at 25% power, the "A" RCP trips.

How will plant parameters associated with the "A" SG respond within the first 20 seconds after the "A" RCP trip?

A. "A" SG Narrow Range level increases.

B. "A" SG Pressure increases.

C. Steam flow from the "A" SG decreases.

D. Primary ΔT across the "A" SG increases.

Proposed Answer: C

Explanation (Optional):

The tripping of the RCP will initially result in decreased flow in that loop, followed by reverse flow in the loop due to DP across the reactor created by the other RCPs. This results in the "A" SG being supplied by T-Cold water, which decreases heat transfer into the "A" SG, initially lowering pressure in the "A" SG ("B" wrong). With lower steam pressure, steam flow decreases ("C" correct) in the affected loop, resulting in a decrease in NR level in that SG due to shrink ("A" wrong). With decreased heat removal in the "A" SG, primary ΔT decreases, ("D" wrong).

Technical Reference(s): Transient Analysis Text TRA510 Figures 13A and B (Rev. 1)
(Attach if not previously provided) Transient Analysis Text TRA510 Table 2 (Rev. 0)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-00147 Describe the effects that a loss or malfunction of a RCP will have of the following systems or plant equipment… Steam Generators… (As available)

Question Source: Bank #75448

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5 and 41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 29	Tier #	2	2
Reactor Coolant Pump:	Group #	1	1
Manually operate / monitor RCP cooling water supplies	K/A #	003.A4.08	
	Importance Rating	3.2	2.9

Proposed Question:

With the plant in MODE 3 at normal operating temperature and pressure, an inadvertent ESF actuation signal occurs, resulting in the following:

- The Reactor Plant Chilled Water containment isolation valves close.
- The RPCCW cross-tie to Chilled Water valves open.

The RO is monitoring the RCPs for proper cooling.

What is the status of cooling to the RCP motors, and is the crew required to trip the RCPs?

A. Cooling to both the motor air coolers and the bearing oil coolers is from RPCCW. The RCPs are not required to be tripped.

B. The motor air coolers have lost cooling while the bearing oil coolers are cooled by RPCCW. The RCPs are not required to be tripped.

C. The motor air coolers have shifted to RPCCW while the bearing oil coolers have lost cooling. The RCPs are required to be tripped.

D. Cooling to both the motor air coolers and the bearing oil coolers has been lost. The RCPs are required to be tripped.

Proposed Answer: B

Explanation (Optional): On a CIA, chilled water (CDS) isolates to CTMT, and the RPCCW cross tie valves to CDS open, supplying neutron shield tank cooling and CAR fans, but not RCP motor cooling ("A" wrong). RPCCW normally supplies the motor bearing oil coolers ("C" and "D" wrong). Since CDS cools the air leaving the RCP motor (assisting in maintaining CTMT temperature), the RCPs are not required to be tripped ("B" correct).

Technical Reference(s): P&ID 121B (Rev. 20)
(Attach if not previously provided) P&ID 122A (Rev. 18) & 122B (Rev. 10)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05427 Describe the following RCP fluid flow paths… RPCCW Flow (As available)

Question Source: Bank #75619

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
Chemical and Volume Control System:	Group #	1	1
Predict the impact of and use procedures to	K/A #	004.A2.12	
mitigate the consequences of a CIA/SIS	Importance Rating	4.1	4.3

Proposed Question:

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

1. An inadvertent Train "B" Safety Injection Signal is received.
2. Train "A" Safety Injection Signal is NOT received.
3. The crew enters E-0 *Reactor Trip or Safety Injection*.

Prior to operator action, what is the status of suction to the Charging Pumps; and how are the operators required by E-0 to specifically address the fact that only one train of SIS actuated?

A. Charging Pump suction is aligned to both the RWST and the VCT. The crew will NOT actuate SIS, to minimize the mass added to the RCS during this inadvertent SIS.

B. Charging Pump suction is aligned to both the RWST and the VCT. The crew WILL actuate SIS to establish a known ECCS system alignment prior to proceeding in the EOP network.

C. Charging Pump suction is aligned to the RWST, and isolated from the VCT. The crew will NOT actuate SIS, to minimize the mass added to the RCS during this inadvertent SIS.

D. Charging Pump suction is aligned to the RWST, and isolated from the VCT. The crew WILL actuate SIS to establish a known ECCS system alignment prior to proceeding in the EOP network.

Proposed Answer: D

Explanation (Optional): On a Safety Injection, the Charging Pump Suction Valves from the RWST (3CHS*LCV112D and E) open (Train-specific). Since these valves are in parallel, suction is aligned. Charging Pump suction from the VCT isolates (3CHS*LCV112B and C) isolate (Train-specific). Since these valves are in series, suction from the VCT has been isolated ("A" and "B" wrong). "A" and "B" are plausible, since pump suction is generally desirable, and only a single train of SI has actuated. "D" is correct, and "C" wrong, since if only a single train of SIS has actuated E-0 directs the crew to actuate the second train of SIS. "C" is plausible, since the SIS is inadvertent, and a single train of SIS will add less mass to the RCS. This action was taken at Salem in response to the eel grass event, and the second train of SIS actuated later in the event.

Technical Reference(s):	E-0 (Rev. 025), step 4.b.RNO
(Attach if not previously provided)	WOG Bkgd Doc (Rev. 2), for E-0 step 4
(including version/revision number)	P&ID 104D (Rev. 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04203 For the below listed pant events, partial or complete, describe the effects on the Chemical and Volume Control System and its interrelated systems… Safety Injection Actuation. (As available)

Question Source: New

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 # 31	Tier #	2	2
Chemical and Volume Control:	Group #	1	1
Predict / monitor changes in parameters associated with	K/A #	004.A1.07	
operating controls for maximum specified letdown flow	Importance Rating	2.7	3.1

Proposed Question:

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

1. The reactor trips.
2. The crew enters ES-0.1 *Reactor Trip Response.*
3. The RO reports letdown has isolated on the trip.
4. After a significant delay, letdown is restored.
5. The crew transitions to FR-I.1 *Response to High Pressurizer Level.*

Upon entry into FR-I.1, conditions are as follows:

- Charging Flow Control Valve 3CHS*FCV121 is in MANUAL.
- Letdown flow is 83 gpm, with letdown orifice isolation valve 3CHS*AV8149B in service.
- Pressurizer level is 90% and stable.
- VCT level is 50% and stable.

In accordance with FR-I.1, the RO places letdown orifice 3CHS*AV8149A in service to lower pressurizer level.

Assuming SIS does not actuate, and no further operator action is taken, what will be the status of the CVCS System thirty minutes after 3CHS*AV8149A was placed in service?

A. VCT level being maintained between 41% and 54%, with an increased makeup frequency.

B. VCT level stable at 66%, with letdown modulate-diverting to Boron Recovery.

C. Letdown isolated, with pressurizer level increasing.

D. Letdown flow exceeding 130 gpm, with the letdown relief valve open.

Proposed Answer: B

Explanation (Optional): The crew has placed the 45 gpm letdown Because letdown is now greater than charging, VCT level will increase, and PZR level will decrease. At 66% VCT level, letdown will modulate-divert to Boron Recovery to maintain VCT level at 66% ("B" correct, "A" wrong). "C" is wrong, since pressurizer level is about 100 gallons / %, so level decreases by about 14% (30 minutes x 45 gallons/minute / 100 gallons/%) during the 30 minutes, which is above the low pressurizer level letdown isolation setpoint of 22%, so letdown has not isolated. "C" is plausible, since PZR level has been decreasing for a half hour, and if the PZR low level letdown isolation setpoint were reached, PZR level would start increasing. "D" is wrong, since, the letdown line pressure control valve will automatically throttle open up to maintain letdown line pressure constant by allowing more flow to the VCT. "D" is plausible, since total flow with the extra orifice in service would be about 130 gpm, and if the relief valve were to open, flow would exceed 130 gpm. Also, when the crew placed the extra orifice in service, pressure would initially increase in the letdown line before the letdown pressure control valve had time to respond.

Technical Reference(s): FR-I.1 (Rev. 008), step 4
(Attach if not previously provided) P&ID 104A (Rev. 49)
(including version/revision number) Functional Drawing 11 (Rev. H)

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: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 32	Tier #	2	2
Residual Heat Removal: Ability to predict and/or monitor	Group #	1	1
changes in heatup/cooldown rates (to prevent exceeding	K/A #	005.A1.01	
design limits) associated with operating RHR controls	Importance Rating	3.5	3.6

Proposed Question:
Initial Conditions:

- RCS Hot Leg Temperature: 330°F.
- RCS Pressure: 360 psia.
- The "B" RCP is running.
- The crew is cooling down the RCS using "A" Train RHR per OP 3208 *Plant Cooldown*.
- 3RHS*FK618 "RHR HDR FLOW" is set to maintain 3,200 gpm flow.
- RHR HX Outlet Valve 3RHS*FCV606 is fully open.

The RO starts to slowly adjust the output of 3RHS*FK618 "RHR HDR FLOW" to 4,000 gpm.

What is the current administrative cooldown rate limit, and how will the RO's actions affect the RCS Cooldown Rate?

	Admin Cooldown Rate Limit	RCS Cooldown Rate
A.	60°F	Increases
B.	60°F	Decreases
C.	75°F	Increases
D.	75°F	Decreases

Proposed Answer: D

Explanation (Optional): The evolution described above is performed when restoring flow to normal after the crew has throttled FCV 618 to increase the RCS cooldown rate. The flowrate measured by 3RHS*FK618 is total flow, through and around the RHR Heat Exchanger. The crew is not adjusting the heat exchanger flow control valve 3RHS*FCV606, which is in the full open position. By increasing RHR Header Flow via controller 3RHS*FK618, more flow is routed through heat exchanger bypass valve 3RHS*FCV618. This lowers RHR pump discharge pressure, decreasing flow through the RHR Heat Exchanger. This provides less cooling to the RCS, decreasing the cooldown rate ("A" and "C" wrong). The administrative cooldown rate limit is 75°F ("B" wrong and "D" correct). "A" and "C" are plausible, since total flow is being increased. "B" is plausible, since 60°F is below the Tech Spec cooldown rate limit, and 60°F is the cooldown rate limit if an RCP has lost seal cooling. Note that Millstone 3 has an RO objective requiring ROs to know the Tech Spec and Administrative reactor vessel (RCS) heatup and cooldown limits.

Technical Reference(s): OP 3208 (Rev. 021-03), section 4.3.12
(Attach if not previously provided) P&ID 112A (Rev. 47)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05445 Describe the operation of the following residual heat removal (RHR) system equipment controls and interlocks… mini-flow control valves… bypass flow control valves… heat exchanger flow control valves… (As available)

MC—05503 State the Technical Specification and Administrative limits for both heatup and cooldown of the reactor vessel.

Question Source: New
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.7, and 41.8
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 33	Tier #	2	2
Emergency Core Cooling:	Group #	1	1
Operational implications of thermodynamics,	K/A #	006.K5.09	
subcooling, superheat, and saturation	Importance Rating	3.3	3.6

Proposed Question:
The following sequence of events occurs:

1. A small cold leg break loss of coolant accident occurs.
2. Safety Injection actuates.
3. The pressurizer empties.
4. RCS pressure decreases to approximately 1200 psia within 2 minutes, and stabilizes.
5. Subcooling based on core exit thermocouples is 0°F.
6. SG pressure is being maintained by the atmospheric relief valves.

Why has pressure stabilized in the RCS?

A. Mass injected into the RCS from the Charging and SIH pumps equals the mass loss from the LOCA.

B. Energy input to the RCS has decreased, since active fuel has started to uncover. Secondary relief valves have throttled closed to maintain the overall heat balance.

C. Energy removal from the RCS via the Steam Generators has stopped. All decay heat from the reactor is being removed by the injection of cold RWST water via CHS and SIH pumps, and by break flow out of the RCS.

D. Break flow is not removing all of the decay heat, and the steam bubble in the reactor vessel is holding up RCS pressure. Excess core heat that isn't being removed out the break is being removed by the SGs.

Proposed Answer: D

Explanation (Optional): "D" is correct, since on a small cold leg break, decay heat exceeds break heat removal, since only liquid is exiting the break. RCS pressure stabilizes at a pressure slightly elevated above SG pressure, with a steam bubble in the vessel, and excess heat being removed by the SG steam dumps. Even with equilibrium RCS pressure, break flow remains in excess of ECCS flow, continuing inventor loss. The top portion of the fuel may eventually uncover. "A" is wrong, since pressure rapidly dropped to 1200 psia and CETCs indicate RCS is at saturation. "A" is plausible, since, for certain small breaks, such as hot leg breaks, pressure decreases to the point where mass in equals mass out, but the pressurizer empties much more slowly. "B" is wrong, since subcooling is zero, and core uncovery is indicated by superheat. "B" is plausible, since a portion of the active fuel may uncover on a small break LOCA, and as fuel uncovers, heat transfer would decrease, requiring secondary relief valves to remove less heat. "C" is wrong, since RCS pressure is slightly above SG pressure. "C" is plausible, since on medium-sized breaks, all heat removal occurs out the break, but RCS pressure in these cases drops below SG pressure.

Technical Reference(s): Westinghouse MITCORE Core Cooling Text (1991), pages 2-9 and 2-10
(Attach if not previously provided) Westinghouse MITCORE Core Cooling Text (1991), Figures 2-2.1 and 2-2.2
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04931 DESCRIBE the events that create pressure equilibrium during a small-break LOCA (As available)

Question Source: Bank #65049
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 # 34	Tier #	2	2
Emergency Core Cooling: Effect of a loss or	Group #	1	1
malfunction on cooling water will have on ECCS	K/A #	006.K6.05	
	Importance Rating	3.0	3.5

Proposed Question:

With the plant in MODE 5, the following sequence of events occurs:

1. All AC Power is lost, and the crew enters EOP 3501 *Loss of All AC Power (MODES 5, 6 And Zero)*.
2. AC Power is restored to the "B" Train from the SBO Diesel.
3. The crew is attempting to establish injection flow using the "B" SIH Pump.
4. The RO reports Service Water cooling for the "B" SI Pump Cooling (CCI) Pump is NOT available.

What effect does the loss of SI Pump Cooling have on the event?

A. The crew will NOT start the "B" SIH Pump. They will establish a gravity feed path to the RCS from the RWST.

B. The crew will NOT start the "B" SIH Pump. They will establish SI Accumulator injection into the RCS.

C. The crew will initiate CCI system feed and bleed cooling from either the Fire Water or Domestic Water Systems, and start the "B" SIH Pump.

D. The crew will align either Fire Water or RPCCW to the Service Water side of the "B" CCI Heat Exchanger, and start the "B" SIH Pump.

Proposed Answer: C

Explanation (Optional): "C" is correct, since if normal cooling for the CCI System is not available, the crew is directed to initiate CCI system feed and bleed cooling from either the Fire Water or Domestic Water Systems ("D" wrong), and start the "B" SIH Pump ("A" and "B" wrong). "A" and "B" are plausible, since cooling has been lost to the "B" SI Pump, and both the gravity feed path and SI Accumulator injection are contingency actions attempted later in EOP 3501. "D" is plausible, since supplying cooling water to the Service Water side of the CCE heat exchanger is the method of alternate cooling used for the Charging Pumps.

Technical Reference(s): EOP 3501 (Rev. 014-01), step 15
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06270 Describe the major action categories within EOP 3501, Loss of All AC Power (Mode 5, 6 and Zero) (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 35	Tier #	2	2
Pressurizer Relief/Quench Tank:	Group #	1	1
Knowledge of abnormal condition procedures	K/A #	007.GEN.2.4.11	
	Importance Rating	4.0	4.2

Proposed Question:

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

1. Pressurizer pressure and level start to decrease.
2. The crew enters AOP 3555 *Reactor Coolant System Leak.*
3. The crew is able to stabilize Pressurizer pressure and level.
4. The RO reports PRT level and temperature are increasing at an abnormal rate.
5. The crew commences taking corrective actions per AOP 3555, Attachment C, "Determination of Leakage to PRT."

Which of the following potential leak paths will Attachment C direct the operators to check?

A. RCS Loop 4 T_H Stop Leakoff Isolation 3DGS-V920 (locally), and PZR Spray valve Leakoff Isolation 3DGS-V910 (locally).

B. Rector Vessel Head Vent temperatures (MB3), and RHS Pump "A" Suction Header Relief Valve 3RHS*RV8708A (locally).

C. RPCCW Surge Tank level (MB2), and RCP Thermal Barrier Return Line temperatures (Plant Process Computer).

D. Charging Line Flow Control Valve Leakoff Isolation 3DGS-V875 (locally), and SI Pump "A" Discharge Header Relief Valve 3SIH*RV8853A (locally).

Proposed Answer: B

Explanation (Optional): "B" is correct, since AOP 3555 Attachment "C" directs these paths to be checked, since they are potential leakage paths to the PRT. "A" is wrong, but plausible, since these paths are checked by AOP 3555, Attachment D if the leakage was into the CDTT. "C" is wrong, but plausible, since these are checked by AOP 3555 if leakage is indicated into the RPCCW system. "D" is wrong, but plausible, since these paths are checked by AOP 3555, Attachment E if leakage is indicated into the PDTT.

Technical Reference(s): AOP 3555 (Rev. 017-00), step 13, and Attachment C
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03912 Describe the major action categories within AOP 3555. (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.3 and 41.10

Comments:

Examination Outline Cross-reference:		Level	RO	SRO
Question # 36		Tier #	2	2
Component Cooling Water : Knowledge of bus power		Group #	1	1
supplies to the CCW Pump, including emergency backup		K/A #	008.K2.02	
Proposed Question:		Importance Rating	3.0	3.2

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

1. The "A" RPCCW Pump trips, and the crew takes all required actions to initially stabilize the plant per AOP 3561 *Loss of Reactor Plant Component Cooling Water*.
2. The crew is preparing to swap the "C" RPCCW Pump from the "B" Train to the "A" Train.
3. The primary rounds PEO is directed to mechanically shift the "C" RPCCW pump and heat exchanger to the "A" Train.
4. The secondary rounds PEO is directed to electrically shift the "C" RPCCW Pump to the "A" Train.

What action is required by the secondary rounds PEO to electrically align the "C" RPCCW pump to the "A" Train?

A. The "C" RPCCW Pump "A" Train breaker, which is normally installed in the "C" RPCCW pump breaker cubicle in 34C, needs to be racked up. The CCP transfer switch does not need to be operated.

B. Nothing needs to be operated at the breaker cubicles, since the "C" RPCCW Pump "A" Train breaker is normally racked up in its 34C breaker cubicle. The CCP transfer switch needs to be operated to realign the "C" RPCCW pump to the "A" train.

C. The "C" RPCCW pump breaker needs to be moved from the "B" train cubicle in 34D and racked up into the "A" train cubicle in 34C. The CCP transfer switch needs to be operated to realign the "C" RPCCW pump to the "A" train.

D. The "A" RPCCW pump breaker will need to be racked down from its breaker cubicle in 34C and racked up into "C" RPCCW pump breaker cubicle in 34C. The CCP transfer switch needs to be operated to realign the "C" RPCCW pump to the "A" train.

Proposed Answer: C

Explanation (Optional): To perform this shift, the breaker must be moved from the "B" Train cubicle ("A" and "B" wrong), and operate the CCP transfer switch, which prevents cross-tying the two trains through the "C" RPCCW Pump ("C" correct). "D" is wrong, since the swing pump has its own separate breaker. "A" is plausible since the swing pump may be aligned to either train, and the "A" and "B" RPCCW Pumps have their own train-specific breakers. "B" is plausible since the CCP transfer switch must be operated when aligning the swing pump to the other train, and some standby plant equipment has racked up breakers. "D" is plausible, since this is how the swing Charging (CHS) pump breaker is operated.

Technical Reference(s): OP 3330A (Rev. 017-02), sections 1.2 and 4.9
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04154 Describe the operation of the RPCCW System under the following normal, abnormal, or emergency conditions… Shifting Pumps and Heat Exchangers… (As available)

Question Source: Bank # 71204
Question History: Last NRC Exam (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 # 37	Tier #	2	2
Component Cooling Water:	Group #	1	1
Predict the impact of and use procedures to mitigate the	K/A #	008.A2.03	
consequences of high/low CCW temperature	Importance Rating	3.0	3.2

Proposed Question:

Initial Conditions:

- The plant is in MODE 5.
- The "A" Train of RHR is in service in the COOLDOWN mode.

The following sequence of event occurs:

1. The RHR HX A RPCCW OUTLET TEMP HI (MB2C) Annunciator is received.
2. The RO confirms the alarm by checking computer point CCP-T65A, which indicates 158°F.

What automatic action occurs due to the high temperature; and what action will the ARP direct the crew to take to mitigate the consequences of the event?

A. RHR Heat Exchanger Total Flow Control Valve 3RHS-FCV618 fails open. The crew will be directed to throttle closed on 3RHS-HC606 "HX A FLOW" controller to reduce RHR flow through the RHR Heat Exchanger.

B. RHR Heat Exchanger Total Flow Control Valve 3RHS-FCV618 fails open. The crew will be directed to throttle open on 3RHS-HC606 "HX A FLOW" controller to increase cooling flow to the Reactor Coolant System.

C. RHR Heat Exchanger Outlet Flow Control Valve 3RHS-FCV606 fails open. The crew will be directed to throttle closed on 3CCP*FK66A1 "RPCCW HX FLOW" controller, to prevent exceeding RPCCW System temperature limits.

D. RHR Heat Exchanger Outlet Flow Control Valve 3RHS-FCV606 fails open. The crew will be directed to throttle open on 3CCP*FK66A1 "RPCCW HX FLOW" controller, to increase cooling flow to the Reactor Coolant System.

Proposed Answer: A

Explanation (Optional): The RHR heat exchanger RPCCW outlet maximum operating temperature is 145°F for both normal and Safety Grade Cold Shutdown (SGCS) operation. With the "HX A FLOW CONT" switch in the "COOLDOWN" position, 3RHS-FCV618 (which bypasses the RHR Heat Exchanger), will fail open if RPCCW outlet temperature increases to 155°F, in order to minimize the heat input from RHR into the RPCCW System ("C" and "D" wrong). Operators will be directed to adjust 3RHS-HC606, "HX A FLOW," controller closed ("B" wrong) to reduce RHR flow through the RHR heat exchanger as necessary, to further minimize heat input into the RPCCW System ("A" correct), and if desired, ADJUST 3CCP*FK66A1 "RPCCW HX FLOW" controller, to increase flow as necessary (without exceeding flow limit) to reduce the RPCCW temperature at the RHR HX outlet to within design limits ("C" and "D" plausible). "B" and "D" are plausible, since temperature is high, and RCS decay heat removal is always a concern.

Technical Reference(s):	OP 3353.MB2C (Rev. 002-07), 1-4
(Attach if not previously provided)	LSK-27-7H (Rev. 13)
(including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04154 Describe the operation of the Reactor Plant Component Cooling Water System under the following normal, abnormal, or emergency conditions… Plant Cooldown… (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.5 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 38	Tier #	2	2
Pressurizer Pressure Control:	Group #	1	1
Ability to monitor automatic operation of PRT	K/A #	010.A3.01	
temperature and pressure during PORV testing	Importance Rating	3.0	3.2

Proposed Question:

The plant is in MODE 3 with PORV Testing in progress, and the following sequence of events occurs:

1. The RO opens the "A" PORV, and the PORV remains open.
2. PRT temperature and pressure start to increase.
3. The PZR REL TK PRESSURE HI annunciator comes in on MB4.
4. All automatic system responses occur, as designed.

How will the PRT temperature and pressure trends initially respond after this annunciator is received?

A. PRT temperature will continue to increase, but pressure will start to decrease.

B. PRT temperature will continue to increase, and pressure will start to increase at a faster rate.

C. PRT temperature and pressure will both continue to increase at the same rate.

D. PRT temperature and pressure will both start to decrease.

Proposed Answer: B

Explanation (Optional): "B" is correct, since at the high-pressure alarm setpoint, the normally open PRT Vent Valve, 3RCS-PCV469, automatically closes, which allows PRT pressure to increase at a faster rate. "A" is wrong, but plausible, since this would occur if the vent valve automatically opened on high pressure. "C" is wrong, but plausible, since this would occur if no automatic actions occurred. "D" is wrong, but plausible, since this would occur if the PRT rupture disk blew at the high pressure setpoint.

Technical Reference(s): OP 3353.MB4A (Rev. 002-14), 2-4

(Attach if not previously provided)

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05347 Describe the operation of the following Pressurizer Relief Tank System controls and interlocks… Vent Valve RCS-PCV469… (As available)

Question Source: Bank # 80880

Question History: Last NRC Exam (Millstone 3 2007 NRC Exam)

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 39	Tier #	2	2
Reactor Protection:	Group #	12	1
Ability to manually operate/monitor	K/A #	012.A4.04	
bistables, trips, resets, and test switches	Importance Rating	3.3	3.3

Proposed Question:

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

1. Power range channel N41 fails low.
2. The crew enters AOP 3571 *Instrument Failure Response*.
3. The crew is preparing to trip bistables associated with the failed PRNI channel.

How are the N41 HI FLUX and RATE TRIP bistables placed in the TRIP condition?

A. The control power fuses are removed from the Power Range N41 drawer.

B. The instrument power fuses are removed from the Power Range N41 drawer.

C. The Comparator Channel Defeat Switch is taken to the "N41" position at the Comparator and Rate Drawer.

D. An I&C Technician trips the bistables in the Train "A" SSPS Input Bay Cabinet in the Instrument Rack Room.

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since AOP 3571 directs the operators to remove control power fuses from Power Range Channel N41, to generate a deenergized "trip" signal to RPS. "B" is plausible, since removing instrument power would also generate a "trip" signal by failing the NIS bistables to the trip position. "C" is plausible, since the Comparator Channel Defeat switch is operated when a PRNI fails, to remove the failed channel from inputting to a Main Board 4 annunciator. "D" is plausible, since this is the method for placing most RPS bistables in the trip condition.

Technical Reference(s): AOP 3571 (Rev. 009-04), Attachment D, page 6 of 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03976 Describe the major action categories contained within AOP 3571. (As available)

Question Source: Bank #69056

Question History: Millstone 3 2000 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 40	Tier #	2	2
Engineered Safety Features Actuation:	Group #	1	1
Operational implications of the definition of a	K/A #	013.K5.01	
safety train and ESF channel	Importance Rating	2.8	3.2

Proposed Question:
With the plant at 100% power, a significant transient occurs on the "A" SG, causing level to rapidly decrease.

What is the coincidence (minimum ESF Channels to trip, versus total ESF Channels) that must sense a lo-lo level condition to generate a reactor trip signal; and to how many safety trains will the trip signal be sent?

A. $\geq$2 of 3 channels must sense the lo-lo level condition, and the signal will be sent to the A Train of SSPS only.

B. $\geq$2 of 3 channels must sense the lo-lo level condition, and the signal will be sent to the A and B Train of SSPS.

C. $\geq$2 of 4 channels must sense the lo-lo level condition, and the signal will be sent to the A Train of SSPS only.

D. $\geq$2 of 4 channels must sense the lo-lo level condition, and the signal will be sent to the A and B Train of SSPS.

Proposed Answer: D

Explanation (Optional): The coincidence for SG lo-lo level trip is 2 (allowing for one failed channel without a reactor trip) out of 4 signals, since SG level also feeds a control system ("A" and "B" wrong). A trip signal is sent to both trains of RPS, to increase the reliability of the trip ("D" correct, and "C" wrong). "A" and "B" are plausible, since trip signals without control systems (such as CTMT pressure or SG pressure) require 2/3 coincidence. "C" is plausible, since the trip signal is on the "A" SG.

Technical Reference(s): FSAR (Rev. 21.3), Sections 7.2.1.1 (Page 7.2-1) and 7.2.2.3.5 (Page 7.2-31 & 32)
(Attach if not previously provided) Functional Drawings 2 (Rev. N),and 7 (Rev. M)
(including version/revision number) Basic RPS Interface Training Drawing (Rev. 0)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05497 Describe the operation of the RPS under the following normal, abnormal, and emergency conditions… Emergency Safeguards Actuation Signal Initiation (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 41	Tier #	2	2
Engineered Safety Features Actuation:	Group #	1	1
Design features / interlocks which provide for avoidance of PTS	K/A #	013.K4.16	
	Importance Rating	3.8	4.2

Proposed Question:

What is a function of the Steam Generator Steam Outlet Nozzle Assembly?

A. Minimize the flowrate/differential pressure against which the MSIVs must open.

B. Prevent backflow from the other SGs, maximizing the cooldown capability of the unaffected SGs.

C. Limit the rate of heat removal from the Reactor Coolant System on a steamline break.

D. Remove moisture from the steam to protect the High Pressure Turbine blades.

Proposed Answer: C

Explanation (Optional): "A", "B", and "D" are wrong, since the purposes of the SG outlet nozzle are:

1. Limit a rapid rise in CTMT pressure on a steamline break.
2. Limit the rate of heat removal from the RCS on a steamline break ("C" correct).
3. Limit thrust forces on main steamline piping on a steamline break.
4. Limit stresses on the SG tubes/tube sheets on a steamline break.

"A" is plausible, since the outlet flow nozzle does limit flow. "B" is plausible, since the flow nozzle will limit flow, and this is a basis for isolating feedwater to a faulted SG. "D" is plausible, since this is the purpose of the moisture separators, which are also in the upper SG area.

Technical Reference(s): FSAR (Rev. 21.3), Section 5.4.4.1, pages 5.4.21 and 22.
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05649 Describe the function… of the following major (Steam Generator) components… Outlet Flow Restrictor… (As available)

Question Source: Bank #69418

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 # 42	Tier #	2	2
Containment Cooling:	Group #	1	1
Predict / monitor changes in Ctmt pressure associated	K/A #	022.A1.02	
with operating Ctmt cooling controls	Importance Rating	3.6	3.8

Proposed Question:

With the plant at 100% power, the crew is preparing to shift to the standby CDS Chiller, and the following sequence of events occurs:

1. The RO stops the "A" CDS Chiller.
2. The crew realigns the appropriate CDS and CCP valves.
3. The RO starts the standby "B" CDS Chiller.
4. The "B" CDS chiller trips due to an RPCCW flow transient, and commences its 30 minute anti-recycle timing sequence.
5. The crew is realigning to place the "A" Chiller back in service.

What is the first operational challenge that the crew will face?

A. Reaching a CTMT temperature Technical Specification limit.

B Reaching a CTMT pressure Technical Specification limit.

C. Reaching an over-temperature condition in the neutron shield tank.

D. Reaching an EEQ temperature limit in the MCC/Rod Control Area.

Proposed Answer: B

Explanation (Optional): The CDS Chillers are 50% capacity, so when only one is running, Reactor Plant Chilled Water (CDS) heats up. As CDS heats up, CTMT temperature increases ("A" plausible), which raises CTMT pressure. "B" is correct, and "A" wrong, since CTMT pressure has much less margin than CTMT temperature before Tech Specs require action. "C" and "D" are wrong, since the neutron shield tank and MCC/RCA temperatures to not pose an immediate concern on loss of one CDS chiller. Millstone 3 chiller trip events confirm this answer. "C" and "D" are plausible, since these loads are cooled by Reactor Plant Chilled Water System.

Technical Reference(s): OP 3330C (Rev. 009-02), section 4.3.2
(Attach if not previously provided) Millstone 3 CR M3-99-2843
(including version/revision number) P&ID 122A (Rev. 18), and 122B (Rev. 10)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04189 Given a failure, partial or complete, of the reactor plant chilled water system, determine effects on the system and on interrelated systems. (As available)

Question Source: Bank #73616

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 # 43	Tier #	2	2
Containment Spray:	Group #	1	1
Ability to monitor automatic operation of Containment	K/A #	026.A3.01	
Spray pump starts and MOV positioning	Importance Rating	4.3	4.5

Proposed Question:
With the plant initially at 100% power, a large-break LOCA occurs, and the following sequence of events occurs:

T + 2 minutes: Containment pressure is 23 psia and increasing.
T + 20 minutes: The crew is preparing to transition from E-0 *Reactor Trip or Safety Injection* to FR-Z.1 *Response to High Containment Pressure.*
T + 20 minutes: In preparation for the transition brief, the RO is comparing the current status of the CTMT Spray System components to their status prior to the event.
T + 20 minutes: The RO observes that both Quench Spray Pumps (3QSS*P3A and B) have started.

What changes (between their initial 100% power status and their current status) should the RO expect as he verifies the status of the following CTMT Spray pumps and valves?

	RSS Pumps (3RSS*P1A-D)	RSS Discharge Valves (3RSS*MOV20A-D)	QSS Discharge Valves (3QSS*MOV34A/B)
A.	Started	Remained Open	Remained Open
B.	Started	Stroked from Closed to Open	Stroked from Closed to Open
C.	Remained Off	Remained Open	Stroked from Closed to Open
D.	Remained Off	Stroked from Closed to Open	Remained Open

Proposed Answer: C

Explanation (Optional): On a CDA (23 psia Ctmt pressure) the QSS Pumps start and their discharge MOVs stroke open from the closed position ("A" and "D" wrong). The RSS Pumps will not start on a CDA until the RWST Lo-Lo setpoint is reached, about 35-40 minutes into the event ("A" and "B" wrong). The RSS discharge valves are normally open ("C" correct). "A" and "B" are plausible, since this is a recent Plant Modification (Fall, 2008). Previously, the RSS pumps started 11 minutes after the CDA. "D" is plausible, since RSS valves remain open, and QSS valves stroke open.

Technical Reference(s): OP 3353.MB2B (Rev. 003-02), 1-8
(Attach if not previously provided) P&IDs 112C (Rev. 38) and 115A (Rev. 36)
(including version/revision number) LSKs 24-9.4A (Rev. 12), 24-9.4B (Rev. 12), 24-9.4Q (Rev. 9), and 27-11J (Rev. 11)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04171 Describe the operation of the following containment de-pressurization system components controls and interlocks… Quench spray system (QSS)… Recirculation spray system (RSS)… (As available)

Question Source: New
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 44	Tier #	2	2
Main and Reheat Steam:	Group #	1	1
Effect of a loss or malfunction of MSS on steam dumps	K/A #	039.K3.06	
	Importance Rating	2.8	3.1

Proposed Question:

With reactor power at 19% and a plant startup in progress per OP 3203 *Plant Startup*, the following sequence of events occurs:

1. The BOP operator closes the main generator output breaker.
2. Turbine load is increased to the point where the steam dump valves have just closed.
3. No other actions have been taken.

Turbine first stage pressure transmitter 3MSS-PT505 (the selected channel on MB7) fails high.

What effect, if any, does this failure have on the steam dump system?

A. All steam dump valves will remain closed. Should actual main steam header pressure increase, they will still respond as needed.

B. All steam dumps throttle fully open, cooling down the RCS to 553°F, at which point all steam dumps close.

C. All steam dumps trip fully open, cooling down the RCS to 553°F, at which point all steam dumps close.

D. All steam dumps throttle fully open, and remain open, resulting in a Low Pressurizer Pressure reactor trip and Low Pressurizer Pressure SI.

Proposed Answer: A

Explanation (Optional): "A" is correct, since the Steam dumps are in the steam pressure mode per OP 3203, step 4.3.1, which means they are already armed, but are responding to 3MSS-PT507 ("D" wrong). "B" and "C" are wrong, but plausible, since this would be the response if 3MSS-PT507 failed high. The valves tripping open versus throttling open is plausible, since the trip-open feature is in service in the Tave mode, but not in the steam pressure mode. "D" is plausible, since if the steam dumps failed open at low power, they would rapidly lower RCS pressure.

Technical Reference(s): OP 3203 (Rev. 019-05), steps 4.3.1 and 4.3.2
(Attach if not previously provided) Functional Dwg 10 (Rev. J)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05006 Given a failure, partial or complete, of the Main Steam System, determine the effects on the system and on interrelated systems. (As available)

Question Source: Modified Bank #75481 Parent Question Attached

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments: Original question #75481 is attached on the next page

Examination Outline Cross-reference:	Level	RO	SRO
Question # 45	Tier #	2	2
Main Feedwater:	Group #	1	1
Ability to recognize system parameters	K/A #	059.GEN.2.2.42	
that are entry conditions for Tech Specs	Importance Rating	3.9	4.6

Proposed Question:

The plant is at 1% power, and the following conditions exist:

- The PRESSURIZER PRESSURE DEVIATION annunciator is lit on MB4.
- The RO reports Pressurizer pressure is 2225 psia and stable.
- Total seal injection flow is 39 gpm and stable.
- The STOP VALVE C ACCUMULATOR PRESSURE LO annunciator is lit on MB5.
- A PEO reports the "C" Feed CTV (3FWS*CTV41C) accumulator pressure is 4700 psig and stable.
- The "B" Feedwater Isolation Valve (3FWS*MOV35B) is OPEN.

Which of these conditions requires entry into a Technical Specification LCO?

A. Pressurizer pressure requires entry into LCO 3.2.5 "DNB Parameters."

B. Seal injection flow requires entry into LCO 3.4.6.2 "Operational Leakage."

C. Feed CTV Accumulator pressure requires entry into LCO 3.6.3 "CTMT Isolation Valves."

D. Feedwater Isolation Valve position requires entry into LCO 3.3.2 "ESFAS Instrumentation."

Proposed Answer: C

Explanation (Optional): "C" is correct, since Feed Accumulators are used to rapidly close the Feed CTVs to isolate CTMT on an accident, and pressure is below the minimum pressure required to satisfy the LCO is 4750 psig. "A" is wrong, since DNB parameters requires Pzr pressure to be above 2204 psia, and it is. "A" is plausible, since pressure is low. "B" is wrong, since the limit for controlled leakage is 40 gpm, but plausible, since total Seal Injection flow is part of the Controlled Leakage Tech Spec. "D" is wrong, since the feedwater isolation MOVs do not receive a FWI input. "D" is plausible, since these valves isolate the Feed flow path through the Feed Reg Valves to CTMT.

Technical Reference(s):	SP 3670.1-009 (Rev. 005-06), page 31 of 57
(Attach if not previously provided)	TRM (Dec. 19, 2003), Table 3.6.3-2, page 3/4.6-14
(including version/revision number)	COLR (Oct 22, 2008), Section 2.13
	LCO 3.4.6.2 (Amendment 238), "Operational Leakage"
	Functional Drawing 13 (Rev. H)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05780 Describe items covered by Technical Specifications. (As available)

Question Source: New

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 # 46	Tier #	2	2
Main Feedwater:	Group #	1	1
Physical connections / cause-effect relationship	K/A #	059.K1.05	
between MFW and RCS	Importance Rating	3.1	3.2

Proposed Question:

During a significant feed heater level transient, HI-HI levels are received in all 3 first point feed heaters and extraction steam responds as designed. The following indications exist in the control room:

- Electrical output has dropped by 15 MWe.
- NIS power has increased to 102%.
- An OTΔT runback commences.

How has the change in feedwater temperature impacted the RCS to result in these NIS and/or ΔT indications?

A. The change in feedwater temperature results in a higher Tave. This causes ΔT to indicate higher than actual power, resulting in an unwarranted OTΔT runback.

B. The change in feedwater temperature results in a higher Tave. This causes ΔT to indicate lower than actual power, potentially preventing a required OTΔT trip.

C. The change in feedwater temperature results in a lower TCold. This adds positive reactivity to the core, and causes NIS Power to indicate higher than actual power.

D. The change in feedwater temperature results in a lower TCold. This adds positive reactivity to the core, and causes NIS Power to indicate lower than actual power.

Proposed Answer: D

Explanation (Optional): Colder feedwater drops Tcold, adding positive reactivity. Power increases, resulting in a greater ΔT, higher Tave ("A" and "B" plausible) and higher NIS power. "A" and "B" are wrong, since ΔT is an accurate indication of reactor power, increasing as Tcold decreases and Thot increases due to the power increase. Tcold in the vessel downcomer has a greater effect on neutron leakage than Tave (lesson learned from Comanche Peak event, 1996), so with colder Tcold, less neutron leakage exists ("C" wrong, and "D" correct). This is supported in this question by diverse indications of high power: higher indicated ΔT and an OTΔT runback, which comes in at a higher power than 102%.

Technical Reference(s): NRC Information Notice- 96.41 Commanche Peak event, 1996
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04881 DESCRIBE the major parameter changes associated with increased heat removal by the Secondary System. (As available)

Question Source: Bank #71059

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 # 47	Tier #	2	2
Auxiliary/ Emergency Feedwater:	Group #	1	1
Predict / monitor changes in AFW flow/motor amps	K/A #	061.A1.05	
associated with operating controls	Importance Rating	3.6	3.7

Proposed Question:

Initial Conditions:

1. The plant is in MODE 4.
2. The crew is preparing to feed all four Steam Generators with Motor Driven AFW Pump A (3FWA*P1A).
3. 3FWA*P1A is running.
4. The MDAFW pump discharge cross-tie valves (3FWA*AOV62A and B) are open.
5. The BOP operator is starting to throttle open on the AFW flow control valves to commence feeding each SG.

The BOP operator is directed to monitor MDAFW Pump A flow, to ensure pump runout conditions are not reached.

Which conditions would indicate that the BOP operator is feeding at the maximum allowable rate?

A. AFW Pump flow reaches 140 gpm per SG.

B. AFW Pump flow reaches 160 gpm per SG.

C. AFW Pump flow reaches 180 gpm per SG.

D. AFW Pump flow reaches 200 gpm per SG.

Proposed Answer: A

Explanation (Optional): To prevent runout conditions, the operator must not exceed a pump flow of 605 gpm, including a Recirc flow of 45 gpm. 605 gpm – 45 gpm Recirc = 560 gpm to the SGs. 560 gpm / 4 SGs = 140 gpm per SG ("A" correct, "B", "C", and "D" wrong). "B", "C" and "D" are plausible, since these flowrates are within the normal feed rate of about 250 gpm to each SG if both MDAFW pumps are running.

Technical Reference(s): OP 3322 (Rev. 021-03), caution prior to step 4.8.4
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04637 Describe the major administrative & procedural a precautions & limitations placed on the operation of the Auxiliary Feedwater System, & the basis for each. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 48	Tier #	2	2
Auxiliary/ Emergency Feedwater:	Group #	1	1
Effect of a loss or malfunction of controllers /	K/A #	061.K6.01	
positioners on AFW	Importance Rating	2.5	2.8

Proposed Question:

With the plant at 100% power, the following Initial Conditions exist:

- The "A" MDAFW Pump is in Pull-To-Lock for a lube oil change.
- Corrosion in the TDAFW Pump Steam Supply Throttle (Governor) Valve (3MSS*MCV5) stem area has caused the valve to stick fully open.

The reactor trips due to a loss of offsite power.

Assuming no operator action is taken, what will be the final AFW flowrate?

A. Flow will increase to, and remain at 400 gpm to each of the 4 SGs, since industry experience shows that governor valve failures will not cause the TDAFW pump to reach its Overspeed trip; and the cavitating venturis will not limit flow at normal SG pressures.

B. Flow will increase to, and remain at 300 gpm to each of the 4 SGs, since industry experience shows that governor valve failures will not cause the TDAFW pump to reach its Overspeed trip; and the cavitating venturis will limit flow to 300 gpm to each SG.

C. Flow will be about 250 gpm to the "B" and "C" SGs only, since the TDAFW Pump will trip on Overspeed.

D. Flow will be about 250 gpm to the "B" and "D" SGs only, since the TDAFW Pump will trip on Overspeed.

Proposed Answer: C

Explanation (Optional): Industry Experience shows several TDAFW Pumps have tripped on overspeed due to corrosion in the Governor Valve stem packing area ("A" and "B" wrong). After the TDAFW pump trips on overspeed, only the "B" MDAFW pump will be running, and the "B" MDAFW pump supplies the "B" and "C" SGs ("C" correct, "D" wrong). "A" is plausible, since 400 gpm/SG is the approximate design flow of the TDAFW pump plus one MDAFW pump. "B" is plausible, since the Cavitating Venturis limit flow to about 300 gpm to a faulted SG. "D" is plausible, since a common convention for train related components is "B" and "D" are associated with the "B" Train.

Technical Reference(s): INPO SER 4-95 Summary (Terry Turbine Governor Valve Stem Binding)
(Attach if not previously provided) P&ID 130B (Rev. 40)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04639 Given a failure, partial or complete, of the Auxiliary Feedwater System, determine the effects on the system and on interrelated systems. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 49	Tier #	2	2
AC Electrical Distribution:	Group #	1	1
Knowledge of power supplies to major system loads	K/A #	062 K2.01	
	Importance Rating	3.3	3.4

Proposed Question:

Which of the following plant loads are normally powered from 4160 KV Bus 34D?

A. The Electric Firewater Pump.

B. "B" Control Building Chiller.

C. "B" Screen Wash Pump.

D. "B" Primary Grade Water Pump.

Proposed Answer: B

Explanation (Optional): "B" is correct, since the "B" HVK Chiller is powered from Bus 34D. "A" is wrong, since the Electric Firewater Pump is powered from 34A via a step-down transformer, or 34B, via MCC 32-1Q . "C" and "D" are wrong, since these loads are powered from Bus 34B. "A", "C", and "D" are plausible, since these loads are important loads powered from 4KV busses.

Technical Reference(s): EE-1M (Rev. 40)
(Attach if not previously provided)
(including version/revision number)

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… At power operations… (As available)

Question Source: Bank #68081

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
DC Electrical Distribution:	Group #	1	1
Physical connections / cause-effect relationship between	K/A #	063.K1.02	
DC and AC distribution	Importance Rating	2.7	3.2

Proposed Question:

The plant is initially at 100 % power, with all electrical systems in their normal alignment.

Load Center 32T deenergizes due to an electrical fault, and cannot be reenergized.

Which of the following describes the effects, if any, on the 125 VDC electrical distribution system?

A. No effect. The inverter 1 static switch will transfer VIAC-1 loads to the alternate 480V AC source.

B. Battery 1 begins to discharge, but will stop discharging when swing charger 301A-3, which still has power, is placed in service.

C. Battery 1 begins to discharge, because it is now supplying Battery Bus 301A-1 loads, while Inverter 1 continues to supply VIAC-1 from its normal 480V AC source.

D. Battery 1 begins to discharge, because it is now supplying Battery Bus 301A-1 loads and supplying VIAC-1 via the inverter.

Proposed Answer: D

Explanation (Optional): "D" is correct, since when 32T is lost, power is lost to the rectifier, the battery charger, and the swing battery charger ("B" and "C" wrong), causing battery 301A-1 to pick up the load on the DC bus. Automatic switchover to the alternate AC source only occurs if the output from the inverter is lost ("A" is wrong), so the battery will also supply VIAC 1 via the inverter. "A" is plausible, since this would occur on a loss of the inverter. "B" is plausible, since batteries 301A-2 and 301B-2 have the swing charger powered from a different load center than their normal charger. "C" is plausible, since this would be the lineup if only the battery charger was lost.

Technical Reference(s): EE-1BA (Rev. 29)
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03325 given a failure of the 480 vac distribution system or a portion of the system, determine the effects on the system and on interrelated systems a). Loss of 480 volt load center or MCC on applicable loads… (As available)

Question Source: Bank #75665

Question History: Millstone 3 2001 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7, 41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 51	Tier #	2	2
Emergency Diesel Generator:	Group #	1	1
Knowledge of EDG Control Power power supplies	K/A #	064.K2.03	
	Importance Rating	3.2	3.6

Proposed Question:

Initial Conditions:

- The plant has tripped from 100% power due to a turbine trip.
- The crew is in EOP 35 ES-0.1 *Reactor Trip Response.*

Numerous MB8 annunciators are received, and the BOP operator reports that the Battery 1 DC voltmeter indicates zero volts.

How does/will the loss of DC Bus 301A-1 affect the "A" Emergency Diesel (EDG)?

A. The "A" EDG auto-started as soon as the DC bus was lost.

B. If an LOP occurs, the "A" EDG will not auto-start, but can still be started from MB8.

C. The "A" EDG can only be started locally using the air start valve levers.

D. If an LOP occurs, the EDG will auto-start, but its output breaker will not automatically close.

Proposed Answer: C

Explanation (Optional): Battery 1 supplies Control power to the "A" EDG start circuit, so it will not automatically or manually start, and air start levers must be used to start the EDG ("C" is correct, and "A" and "B" are wrong). "A" and "B" are plausible, since each of these show a problem with the EDG starting circuit. "D" is wrong, but plausible, since this would result if the "A" sequencer was denenergized, but it receives power from vital instrument AC power.

Technical Reference(s): AOP 3563 (Rev. 009-01), Attachment A, step 6, and load list, page 8 of 8.
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03309 Given a failure of the 125 VDC distribution system or a portion of the system, determine the effects on the system and on interrelated systems... (As available)

Question Source: Bank #74348

Question History: Millstone 3 2000 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 52	Tier #	2	2
Process Radiation Monitoring:	Group #	1	1
Effect of a loss or malfunction of Rad Monitors on	K/A #	073.K3.01	
radioactive effluent releases	Importance Rating	3.6	4.2

Proposed Question:

Initial conditions:

1. A small tube leak is in progress.
2. Chemistry/Operations have just determined leakage is within Tech Spec limits.

The Turbine Building Floor Drain Sump Rad Monitor (3DAS-RE50) fails high.

What automatic action will result from the failed radiation monitor?

A. Turbine Building Floor Drains Sump diverts to the TPCCW Sump.

B. Turbine Building Floor Drains Sump diverts to the High Level Waste Drain Tanks.

C. Turbine Building Floor Drains Sump diverts to the Condensate Demin Waste Neutralizing Sump.

D. Turbine Building Floor Drains Sump diverts to the Auxiliary Building Sump.

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since RE50 diverts flow to the TPCCW sump. "B" is plausible, since the TPCCW sump is normally aligned to the Liquid Waste System. "C" is plausible, since Waste Neutralizing Sump Monitor 3CND-07 diverts to the Waste Neutralizing Sump. "D" is plausible, since Aux Condensate Flash Tank Rad Monitor 3CNA-RE47 diverts Aux Condensate to the Aux Bldg Sump.

Technical Reference(s): AOP 3573 (Rev. 018), Attachment A, page 4 of 12.
(Attach if not previously provided) P&ID 106C (Rev. 45)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05293 Describe the operation of the following Radiation Monitors controls and interlocks: A. DAS-RE50. B. CNA-RE47. C. CND-RE07. D. LWS-RE70. E. SSR-RE08 (As available)

Question Source: Bank #67219

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 # 53	Tier #	2	2
Service Water:	Group #	1	1
Monitor automatic operation of Service Water	K/A #	076 A3.02	
emergency heat loads	Importance Rating	3.7	3.7

Proposed Question:

With the plant initially at 100% power, a loss of offsite power occurs (the SAFETY INJECTION ACTUATION Annunciator is **NOT** lit).

The RO is monitoring his boards to verify proper response of Service Water System valves.

Which Service Water System valves will the RO observe changing position, and to what position will they change?

	EDG cooling outlet valves	TPCCW HX supply valves	RPCCW HX supply valves
A.	Strokes open	Remains open	Strokes closed
B.	Remains open	Strokes closed	Strokes closed
C.	Remains open	Remains open	Remains open
D.	Strokes open	Strokes closed	Remains open

Proposed Answer: D

Explanation (Optional): "D" is correct, since on an LOP, the diesel outlet valves stroke open ("B" and "C" wrong), the TPCCW Service Water Supply valves stroke closed ("A" and "C" wrong), and the RPCCW valves remain open ("A" and "B" wrong). "A" and "B" are plausible, since RPCCW valves automatically close on a CDA. "B" and "C" are plausible, since the Emergency Diesel Valves remaining open would be true if they were normally open, as the other heat exchanger valves are. "A" and "C" are plausible, since TPCCW valves remaining open on an SIS.

Technical Reference(s): EM 133B (Rev. 72)
(Attach if not previously provided)
(including version/revision number)

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… Loss of Offsite Power… (As available)

Question Source: Bank #69670

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 # 54	Tier #	2	2
Instrument Air:	Group #	1	1
Knowledge of power supplies to instrument air compressor	K/A #	078.K2.01	
	Importance Rating	2.7	2.9

Proposed Question:

An electrical fault occurs in the "B" Train 4160 volt bus tie breaker, resulting in a reactor trip and a loss of both 34B and 34D.

What will be the status of instrument air (IAS) during the performance of ES-0.1 *Reactor Trip Response*?

A. Both instrument air compressors are still available.

B. The "A" instrument air compressor will be running, but the "B" instrument air compressor has been lost.

C. Both IAS compressors will be lost, but the Service Air compressor will maintain IAS system pressure.

D. Both of the instrument air compressors and the service air compressor have been lost.

Proposed Answer: D

Explanation (Optional): The power supply to the "A" IAS compressor is via 34B ("A" and "B" wrong), and the power supply to the "B" compressor is via 34D. The power supply to the service air compressor is via 34B ("C" wrong, "D" correct). "A" is plausible, since both IAS compressors receive power from the same train. "B" is plausible, since almost all equipment at Millstone 3 is powered from opposite trains. "C" is plausible, since the SAS compressor is not labeled with a train designator.

Technical Reference(s): Form OP 3332A-004 (Rev. 004-03)
(Attach if not previously provided) Form OP 3332C-003 (Rev. 000-01)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05321 Describe the operation of the following plant air systems components... Service Air Compressor... Instrument Air Compressors... (As available)

Question Source: Bank #76280

Question History: Millstone 3 2002 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 # 55	Tier #	2	2
Containment:	Group #	1	1
Physical connections / cause-effect between	K/A #	103.K1.07	
containment and the containment vacuum system	Importance Rating	3.5	3.7

Proposed Question:

An inadequate core cooling event has occurred, and the crew has aligned the Containment Vacuum System to remove hydrogen from Containment.

What is the flowpath between the Containment atmosphere and the Containment Vacuum System during this evolution?

A. Both vacuum pumps are operated via their normal flowpaths to supply and remove air from containment.

B. The vacuum air ejector is aligned to take a suction on containment and exhaust to the site ventilation stack.

C. One vacuum pump is lined up to supply air to containment from the Aux. Bldg, and the other vacuum pump uses the normal exhaust flowpath.

D. One vacuum pump is lined up to supply air to containment from the Aux. Bldg. The air ejector takes a suction on CTMT and exhausts to the site ventilation stack.

Proposed Answer: C

Explanation (Optional): "C" is correct since per OP 3313E, section 7.4, one pump is lined up as a supply pump and the other is used as an exhaust pump. "A" is wrong since it provides no air supply path. "A" is plausible, since this is its normal lineup. "B" and "D" are wrong, since the air ejector is not used. "B" and "D" are plausible, since the air ejector is a possible flowpath from CTMT through the Containment Vacuum System.

Technical Reference(s): OP 3313E (Rev. 008-06), section 4.4
(Attach if not previously provided) P&ID 148E (Rev. 23) and 153A (Rev. 28)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04271 Describe the operation of the Containment Vacuum System under the following normal, abnormal and emergency conditions… Performance of backup containment purge system operations. (As available)

Question Source: Bank #60279

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4 and 41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 56	Tier #	2	2
Reactor Coolant:	Group #	2	2
Predict the impact of and use procedures to mitigate	K/A #	002.A2.02	
the consequences of a loss of coolant pressure	Importance Rating	4.2	4.4

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

1. The PRESSURIZER PRESSURE DEVIATION Annunciator is received on MB4.
2. The RO reports RCS pressure is 2220 psia and decreasing.
3. The RO confirms that all 4 PZR Pressure channels are decreasing.
4. The RO reports PORV Outlet Temperature indicates 110°F on MB4.

At this pressure, what is the expected status of the Pressurizer Backup Heaters, and what action will the Annunciator Response Procedure (ARP) direct?

A. The PZR Backup Heaters should have already energized. The RO will simultaneously close both PORV Block Valves, and energize the Backup Heaters, if needed, from MB4.

B. The PZR Backup Heaters should have already energized. The RO will manually close any failed-open Pzr Spray Valves, and energize the Backup Heaters, if needed, from MB4.

C. The PZR Backup Heaters should not be energized. The RO will simultaneously close both PORV Block Valves from MB4.

D. The PZR Backup Heaters should not be energized. The RO will manually close any failed-open Pzr Spray Valves from MB4.

Proposed Answer: B

Explanation (Optional): "B" is correct, and "A" wrong, since, if the PORVs were leaking, their tail pipe temperatures would be high, bringing in an associated annunciator. "A" is plausible, since a leaky PORV would also lower RCS pressure. "C" and "D" are wrong, since the backup heaters come on when pressure is 25 psi below the normal pressure of 2250 psia). "C" and "D" are plausible; since backup heaters do not energize until PZR pressure is 25 psi below normal.

Technical Reference(s): OP 3353.MB4A (Rev. 002-14), 4-4
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05446 Describe the operation of the Reactor Coolant System under normal, abnormal, and emergency operating conditions. (As available)

Question Source: New

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 # 57	Tier #	2	2
Pressurizer Level Control:	Group #	2	2
Purpose of major system components and controls	K/A #	011.GEN.2.1.28	
	Importance Rating	4.1	4.1

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

1. A turbine runback initiates.
2. During the runback, both PZR spray valves start to throttle open.
3. As the runback continues, the PZR backup heaters energize, even though spray valves are still open.

For what purpose have the backup heaters energized?

A. The PZR level controller is responding to a greater than 5% <u>outsurge</u> from the downpower, to restore the PZR liquid to saturation conditions.

B. The PZR level controller is responding to a greater than 5% <u>insurge</u> from the downpower, to restore the PZR liquid to saturation conditions.

C. The PZR pressure controller is responding to the PZR pressurizer pressure <u>rise</u> via a rate/lag compensated circuit, to prevent pressure oscillations as pressure is restored to 2250 psia.

D. The PZR pressure controller is responding to the PZR pressurizer pressure <u>drop</u> via a rate/lag compensated circuit, to prevent pressure oscillations as pressure is restored to 2250 psia.

Proposed Answer: B

Explanation (Optional): The downpower will cause RCS temperature to increase due to a decrease in heat removal. This will cause RCS water to expand, resulting in a insurge to the pressurizer, so both PZR pressure and level will increase. The increase in pressure causes spray valves to open, and when pressurizer level increases by 5%, the heaters will energize ("B" is correct, "A" is wrong). The reason for this is that the temperature of the insurging water is not as hot as the pressurizer water, and if an outsurge follows with the pressurizer water at less than saturation temperature, RCS pressure could rapidly drop. "C" and "D" are wrong since backup heaters cycle around 2225 to 2233 psia, and spray valves cycle around 2275 to 2325 psia, and shouldn't both be on together based on pressure.

Technical Reference(s): Functional Sheet 11 (Rev. H)
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05341 Describe the operation of the Pressurizer Pressure and Level Control System under Normal, Abnormal, and Emergency Operating conditions. (As available)

Question Source: Bank #68619

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 # 58	Tier #	2	2
Rod Position Indication:	Group #	2	2
Knowledge of LCOs and Safety Limits	K/A #	014.GEN.2.2.22	
	Importance Rating	4.0	4.7

Proposed Question:

With the plant at 100% power, the following annunciator is received:

RPI NON URGENT FAILURE

The RO reports that the DATA B FAILURE light is lit on the DRPI display.

What is the current accuracy of DRPI (indicated DRPI versus actual rod position) for the affected rods, and is DRPI accuracy within Technical Specification limits?

A. The accuracy is +4 and – 12 steps. This is within the Tech Spec limit.

B. The accuracy is +4 and – 10 steps. This is within the Tech Spec limit.

C. The accuracy is +4 and – 12 steps. This is NOT within the Tech Spec limit.

D. The accuracy is +4 and – 10 steps. This is NOT within the Tech Spec limit.

Proposed Answer: B

Explanation (Optional): DRPI normally is accurate to within ±4 steps from actual rod position. On a loss of DATA B, accuracy becomes +4 and – 10 steps ("A" and "C" wrong). "B" is correct, and "D" wrong, since this is within the Tech Spec limit of ±12 steps. "C" and "D" are plausible, since accuracy has decreased with the loss of DATA B. "A" is plausible, since 12 steps is the Tech Spec limit.

Technical Reference(s): OP 3353.MB4C (Rev. 006-00), 3-10
(Attach if not previously provided) AOP 3552 (Rev. 009), Attachment C, step 3.
(including version/revision number) Tech Spec LCO 3.1.3.2 (Amendment 229)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05485 Given one of the below partial or complete failures of the Rod Position Indication System, determine the effects on the system and on inter-related systems.... Data "A" or Data "B" failure... (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5 and 43.2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 59	Tier #	2	2
Nuclear Instrumentation:	Group #	2	2
Effect of a loss or malfunction of sensors on NIS	K/A #	015.K6.01	
	Importance Rating	2.9	3.2

Proposed Question:

Initial Conditions:

- The plant is at 100% power.
- NIS Power Range Channel N41 has failed high.
- All appropriate bistables have been tripped.

A plant shutdown is commenced due to an approaching hurricane.

How will the shutdown be affected if an additional power range channel fails-as-is at its 100% value during the shutdown?

A. Automatic outward rod motion will not be blocked if rod control is still in automatic when power is reduced below 15% power.

B. The reactor will automatically trip when power is reduced below 10% power.

C. Both source range channels will have to be manually energized from MB4.

D. Neither source range channel can be energized from MB4. Gamma-Metrics will be used for Source Range indication.

Proposed Answer: D

Explanation (Optional): "A" is wrong, but plausible, since C-5 (15% power auto rod block) was deleted as part of the stretch power up-rate project, completed in the Fall of 2008. "B" is wrong, but plausible, since power above P-10 (10%) blocks the IR high flux trip and the PR high flux low setpoint trips. "D" is correct, and "C" wrong, but plausible, since P-10 blocks both source ranges from energizing, either automatically or manually.

Technical Reference(s): Functional drawings 3 (Rev. G), 4 (Rev. G), and 16 (Rev. L).
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05229 For the following conditions, determine the effects on the Nuclear Instrumentation System and on interrelated systems… Failure of two or more Power Range channels (As available)

Question Source: Bank #73536

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 60	Tier #	2	2
Non-nuclear Instrumentation:	Group #	2	2
Design features / interlocks which provide for reading	K/A #	016.K4.01	
NNI outside control room	Importance Rating	2.8	2.9

Proposed Question:

The crew has just evacuated the control room due to a fire in the Instrument Rack Room.

Why will the crew realign the Umbilical Cord at the Auxiliary Shutdown Panel (ASP) to the Switchgear Room plug?

A. This removes the Main Board Controllers from the control circuitry, and places the Auxiliary Shutdown Panel Controllers in service.

B. This aligns the Appendix "R" controllers at the Auxiliary Shutdown Panel, and isolates the inputs from automatic actuation signals.

C. This realigns the Wide Range T-Cold indication on the Auxiliary Shutdown Panel from "B" Train power to "A" Train power.

D. This realigns the Train A and Train B ASP instrumentation from the Instrument Rack Room to the Spec 200 instrumentation.

Proposed Answer: C

Explanation (Optional): "A" is wrong, but plausible, since the "Remote/Local" switches at the Transfer Switch Panels select the Aux Shutdown Panel controllers. "B" is wrong, but plausible, since the Fire Transfer Switch Panel "Remote/Isolate" switches remove the MB controllers from the circuit, including automatic actuation signals. "C" is correct, since the ASP umbilical cord transfers the WR T-Cold instrument power from Train "B" to train "A" power. "D" is wrong, but plausible, since the FTSP umbilical cord switches instrumentation from the Main Control plug to the Switchgear Room Spec 200 cabinet plug. This actually switches the indication to an entirely separate circuit.

Technical Reference(s): EOP 3509.1 (Rev. 012) Note prior to step 20, and step 20.b

(Attach if not previously provided) ASP115C (Rev. 1) Training Text, pages 20 and 21

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03294 State the purpose of the Fire Transfer Switch Panel Umbilical Cord and Auxiliary Shutdown Panel Umbilical Cord. (As available)

Question Source: Bank #80195

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 10CFR55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 61	Tier #	2	2
In-Core Temperature Monitor:	Group #	2	2
Operational implications of saturation and subcooling	K/A #	017.K5.02	
	Importance Rating	3.7	4.0

Proposed Question:
Current conditions are as follows:

- A steam bubble exists in the reactor vessel head.
- The crew has entered FR-I.3 *Response to Voids in Reactor Vessel.*
- The "A" Charging Pump is running.
- Pressurizer Pressure: 1100 psia and stable.
- CETC Temperature: 556°F and stable.

In accordance with the WOG Background Document, what is the preferred method for eliminating this steam bubble in the vessel head?

A. Increase RCS pressure using pressurizer heaters.

B. Start two Reactor Coolant pumps.

C. Start two CRDM cooling fans.

D. Vent the reactor vessel head to the PRT.

Proposed Answer: A

Explanation (Optional): "A" is correct, since FR-I.3, step 7 will use Pressurizer heaters to raise pressure 50 psi in an effort to condense steam voids by increasing RCS pressure above the void's fluid saturation point, and this method is most effective for a saturated steam bubble. Steam tables show the pressure/temperature conditions indicate a saturated steam bubble. If the void collapses, the crew exits FR-I.3. If the void fails to collapse, one RCP is started to force cooling into the upper head (most effective if the steam bubble was superheated) from the downcomer and should collapse a steam void ("B" plausible). Only one RCP is run to limit the potential for sweeping non-condensable gases into the rest of the RCS ("B" wrong). FR-I.3 does not operate CRDM cooling fans to collapse the void ("C" wrong), but ES-0.2 Natural Circulation Cooldown operates CRDM fans to prevent void formation by removing heat from the head ("C" plausible). Venting the head is used to vent a non-condensable void as might exist following accumulator injection ("D" wrong, but plausible).

Technical Reference(s): FR-I.3 (Rev. 012), steps 7-9
(Attach if not previously provided) WOG Background (Rev. 2); FR-I.3, step 6
(including version/revision number)

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

Learning Objective: MC-04548 Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 FR-I.3. (As available)

Question Source: Bank #64236

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.5 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 62	Tier #	2	2
Containment Purge:	Group #	2	2
Monitor automatic operation of Ctmt purge isolation	K/A #	029.A3.01	
	Importance Rating	3.8	4.0

Proposed Question:

With the plant in MODE 6, CTMT Radiation Monitor 3RMS*RE41 goes into HI ALARM.

The crew is monitoring to ensure the associated automatic actions occur.

What plant response will the crew observe?

A. The running CTMT Vacuum Pump trips on MB-2.

B. CTMT Isolation Phase "A" actuates on MB-2.

C. The running CTMT Purge Fan and Air Handling Unit trip on VP-1.

D. The CTMT Purge Supply and Exhaust Valves close on VP-1.

Proposed Answer: D

Explanation (Optional): "D" is correct, and "A", "B", and "C" wrong, since High Radiation closes CTMT Purge Supply and Exhaust Dampers. "A", "B", and "C" are plausible, since each of these would minimize the spread of radiation, and their monitoring location is correct.

Technical Reference(s): AOP 3573 (Rev. 018), Attachment B, page 6 of 6.
(Attach if not previously provided) P&ID 153A (Rev. 28)
(including version/revision number)

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… RMS-RE-41/42… (As available)

Question Source: Bank #64315

Question History: Millstone 3 2000 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 10CFR55.41.7 and 41.11

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 63	Tier #	2	2
Spent Fuel Pool Cooling:	Group #	2	2
Knowledge of surveillance procedures	K/A #	033.GEN.2.2.12	
	Importance Rating	3.7	4.1

Proposed Question:

In accordance with SP 3670.1 *Control Room and PEO Surveillances*, what two items related to the spent fuel pool are checked by the Control Room operators on a shiftly or weekly basis?

A. Fuel Pool level is verified above its low-level alarm setpoint weekly by checking the Fuel Pool Low Level annunciator extinguished on MB1. Fuel Pool Purification is checked weekly by checking one Spent Fuel Pool Purification Pump running on MB1.

B. Fuel Pool level is verified above its low-level alarm setpoint weekly by checking the Fuel Pool Low Level annunciator extinguished on MB1. Fuel Pool temperature is recorded shiftly from the Spent Fuel Pool Temperature Indicator on MB1.

C. Fuel Pool Cooling is checked daily by checking one Spent Fuel Pool Cooling Pump running on MB1. Fuel Pool Purification is checked weekly by checking one Spent Fuel Pool Purification Pump running on MB1.

D. Fuel Pool Cooling is checked daily by checking one Spent Fuel Pool Cooling Pump running on MB1. Fuel Pool temperature is recorded shiftly from the Spent Fuel Pool Temperature Indicator on MB1.

Proposed Answer: B

Explanation (Optional): "B" is correct, since Fuel Pool level is verified above its low-level alarm setpoint weekly by checking the Fuel Pool Low Level annunciator extinguished on MB1, and Fuel Pool Temperature is recorded shiftly from the Spent Fuel Pool temperature indicator on MB1. "A", "C", and "D" are wrong, since Spent Fuel Pool pumps are not required to be verified running by surveillance. "A", "C", and "D" are plausible, since fuel pool cooling and purification are important, and fuel pool cooling pump control switches are located at MB1. Also, Fuel Pool Cooling Pumps are surveilled for vibration and flow checks.

Technical Reference(s): SP 3670.1-001 (Rev. 026-01), page 3.
(Attach if not previously provided) SP 3670.1-002 (Rev. 011-02), page 3.
(including version/revision number) P&ID 111A (Rev. 32)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05642 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 Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 64	Tier #	2	2
Steam Generator:	Group #	2	2
Physical connections / cause-effect relationship between	K/A #	035.K1.01	
S/Gs and MFW/AFW	Importance Rating	4.2	4.5

Proposed Question:

Current Conditions:

- The crew is conducting a plant startup using OP 3203 *Plant Startup*.
- Turbine load is 160 MWe and slowly increasing.
- Steam dump demand is 8% and slowly decreasing.

At this power level, what is the feedwater flowpath from the exit of the First Point Feedwater Heaters to the "A" Steam Generator?

A. Through FW Isolation Valve 3FWS-MOV35A, to Main Feedwater Regulating Valve 3FWS*FCV510, to the FW Isolation Trip Valve 3FWS*CTV41A, and into the SG.

B. Through Main Feedwater Regulating Valve 3FWS*FCV510, to FW Isolation Trip Valve 3FWS*CTV41A, and into the SG.

C. Through FW Isolation Valve 3FWS-MOV35A to Main Feedwater Regulating Bypass Valve 3FWS*LV550, to FW Isolation Trip Valve 3FWS*CTV41A, and into the SG.

D. Through Main Feedwater Regulating Bypass Valve 3FWS*LV550 to FW Isolation Trip Valve 3FWS*CTV41A, and into the SG.

Proposed Answer: D

Explanation (Optional): Power is below the switchover level indicated by steam dumps still open and therefore, flow is still through the bypass valves. Caution in 3203 instructs the operator not to switch to the main feed regulating valves until the steam dumps are closed ("A" and "B" wrong). "D" is correct, and "C" wrong, since flow through the bypass valve is around the isolation valve, to the CTV then into the SG.

Technical Reference(s): OP3203 (Rev. 019-05), steps 4.2.2.e, 4.2.4, 4.3.56, and 4.3.57
(Attach if not previously provided) OP 3321 (Rev. 017), steps 4.5.3 to 4.5.6
(including version/revision number) P&ID EM-130C (Rev. 24)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04658 Given a Main Feedwater System Diagram, DESCRIBE the system flowpath and electrical alignment under the following normal, abnormal, & emergency operating conditions… Plant Startup (Feed Regulating Bypass Valves controlling)… (As available)

Question Source: Bank #69867

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 # 65	Tier #	2	2
Steam Dump/ Turbine Bypass Control:	Group #	2	2
Knowledge of annunciators, indications, or response	K/A #	041.GEN. 2.4.31	
procedures	Importance Rating	4.2	4.1

Proposed Question:

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

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

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

A. HI T ERROR T AVE – T REF C-16

B. TURBINE BYPASS VV TRIPPED OPEN

C. TURB BYPASS VV ARM FOR OPENING

D. TURB BYPASS T AVE INTLK BYPASSED

Proposed Answer: C

Explanation (Optional): "C" is correct, since PT 506 failing low will generate a C-7 Load Reject signal, arming the dumps; and selecting RESET removes the arming signal. "A" is wrong, put plausible, since this alarm would come in if Tave is 20°F below Tref, and Tref will change on the selected PT505 or 506 failure. "B" is wrong, but plausible, since this is driven by PT505, and comes in if Tref is significantly below Tave. "D" is wrong, but plausible, since this annunciator comes in on Low Tave when the Interlock Selector Switch is selected to Bypass with low Tave.

Technical Reference(s): AOP 3571 (Rev. 009-04), Attachment G, page 1 of 3
(Attach if not previously provided) LSK 3-1.1C (Rev. 7), and 3-1.1E (Rev. 7)
(including version/revision number) Functional Drawing 10 (Rev. J)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05630 Describe the operation of the following steam dump system controls and interlocks: Steam Dump Mode Selector Switch… C-7 Interlock… (As available)

Question Source: New

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 # 66	Tier #	3	3
Knowledge of requirements for controlling	Group #	1	1
vital/controlled access	K/A #	GEN.2.1.13	
Proposed Question:	Importance Rating	2.5	3.2

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

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

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

A. Both escorts are required to remain with the visitors and escort them outside the protected area prior to the one escort leaving for the control room.

B. The one remaining escort may take escort responsibility for all 10 visitors and remain outside of the switchgear room until the second escort returns.

C. The one remaining escort may take 5 visitors into the switchgear area, while the other escort takes the other 5 visitors with him into the control room.

D. The one remaining escort may take escort responsibility for all 10 visitors and continue the tour into the switchgear area.

Proposed Answer: B

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

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

Proposed references to be provided to applicants during examination: None

Learning Objective: GE-00177 Describe escorting responsibilities. (As available)

Question Source: Bank #64349

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 # 67	Tier #	3	3
Knowledge of criteria requiring a plant-wide announcement	Group #	1	1
	K/A #	GEN.2.1.14	
	Importance Rating	3.1	3.1

Proposed Question:
With the plant at 100% power, the following evolutions occur over the course of the shift:

1. Radiography is commenced in the Auxiliary Building CTMT penetration area.
2. A turbine building exhaust fan is started.
3. A PEO locally swaps condensate demineralizers.
4. The "C" Condensate Pump trips; and the crew manually starts the standby pump.

In accordance with OP-AA-100 *Conduct of Operations*, which of these evolutions specifically required the crew to make a plant announcement to alert personnel of changing conditions?

A. The changing radiological conditions in the Aux Bldg.

B. The start of the turbine building exhaust fan.

C. The swap of condensate demineralizers.

D. The unplanned start of the standby condensate pump.

Proposed Answer: A

Explanation (Optional): Operations personnel are required to announce the following:
Starting or stopping plant equipment (Large 480 volt loads or greater) ("B", "C", and "D" plausible).

- Major plant equipment (large 480 volt loads or greater operated from the Control Room ("C" wrong). It is not expected that changes in running status be announced for minor loads such as process radiation monitor fans, sump pumps, Turbine Building vent fans ("B" wrong), etc.
- When starting or stopping plant equipment, Operations personnel will announce the planned activity ("D" wrong, since the condensate pump start was not planned), with direction for plant personnel to stand clear of the equipment.

Changing radiological conditions
Operations personnel will announce the planned activity with direction that personnel stand clear of areas potentially impacted ("A" correct). A subsequent announcement will be made when normal area access is restored.

Technical Reference(s): OP-AA-100 (Rev. 6), Attachment 2, Section 16
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06578 Outline duties and responsibilities of the Control Room Operator. (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 68	Tier #	3	3
Ability to interpret and execute procedure steps	Group #	1	1
	K/A #	GEN.2.1.20	
	Importance Rating	4.6	4.6

Proposed Question:

In accordance with DNAP 0509 *Dominion Nuclear Procedure Adherence and Usage*, what specific requirements exist while executing steps from a "Reference Use" procedure?

A. The operator will keep the procedure readily available where the work activity is being performed, will refer to the procedure as often as necessary to ensure proper performance of the activity.

B. The operator will keep the procedure in hand for steps marked as "critical steps," and keep the procedure readily available, though not necessarily at the work location, for the remaining steps.

C. The operator will keep the procedure in hand, will read each step directly before performance of that step, and will notate each step as it is completed.

D. The operator will keep the procedure available, though not necessarily at the work location, and can perform the steps without referring to the procedure; however, the user is still responsible for adhering to the procedure.

Proposed Answer: A

Explanation (Optional): "A" is correct, since a Reference Use procedure shall be used as follows:
1. The procedure shall be readily available to the user.
2. The procedure shall be referenced prior to job start and as often as necessary to ensure proper performance of the activity.
"B" is wrong, but plausible, since this describes the requirements of a "Multiple Use" procedure.
"C" is wrong, but plausible, since this describes the requirements of a "Continuous Use" procedure.
"D" is wrong, but plausible, since this describes the requirements of an "Information Use" procedure.

Technical Reference(s): DNAP-0509 (Rev. 11), Sections 3.2.6 to 3.2.8
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06762 State the criteria that would assign a procedure as a Continuous Level of Use (As available)

Question Source: New

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	
Ability to determine expected plant configuration	Group #	2	
using documents	K/A #	GEN.2.2.15	
Proposed Question:	Importance Rating	3.9	4.3

Current plant conditions:

1. An operator has completed the initial positioning (Valve Lineup) on the Train A High Pressure Safety Injection System valves.
2. A second operator has just been dispatched to perform a Valve Position Verification on the system.
3. Two of the valves the operator is verifying are:
   - 3SIH*V89 (3SIH*P1A DIS PRES INST PI919 ISOL)
   - 3SIH*V107 (3SIH*P1A TO HOT LEG #2 BALANCING)

Using the attached copy of OP 3308-003 "Train A High Pressure Safety Injection System," how is the operator required to verify the positions of these two valves?

A. The operator will turn the hand wheel for 3SIH*V89 in the CLOSED direction until stem movement is verified, then turn the hand wheel until the valve is fully open, and then CLOSE the valve ¼ turn. The operator will check the locking device secured for 3SIH*V107.

B. The operator will turn the hand wheel for 3SIH*V89 in the CLOSED direction until stem movement is verified, then turn the hand wheel until the valve is fully open, and then CLOSE the valve ¼ turn. The operator will remove the locking device for 3SIH*V107, CLOSE the valve, OPEN the valve the required number of turns and reinstall the locking device.

C. The operator will turn the hand wheel for 3SIH*V89 in the CLOSED direction until stem movement is verified, and then turn the hand wheel until the valve is fully open. The operator will check the locking device secured for 3SIH*V107.

D. The operator will turn the hand wheel for 3SIH*V89 in the CLOSED direction until stem movement is verified, and then turn the hand wheel until the valve is fully open. The operator will remove the locking device for 3SIH*V107, CLOSE the valve, OPEN the valve the required number of turns and reinstall the locking device.

Proposed Answer: C

Explanation (Optional): BOP indicates 3SIH*V89 is to be Back-seated Open. This is accomplished by turning the hand wheel in the CLOSED direction until stem movement is verified, and then turning the hand wheel until the valve is fully open ("A" and "B" wrong). LT indicates 3SIH*V107 is to be Locked Throttled. This is accomplished by checking the locking device secured. This valve should not be adjusted, since it was set based on flow per the surveillance procedure ("C" correct, "D" wrong). "A" and "B" are plausible; since the normal method for checking valves open is to close the valve ¼ turn off the open seat. "D" is plausible, since this method is normally used to initially position throttled valves.

Technical Reference(s): OP 3308-003 (Rev. 004-04)
(Attach if not previously provided) MP-14-OPS-GDL601 (Rev. 000-02), sections 3.1, 3.7.4, 3.8.1, and 3.8.4
(including version/revision number)

Proposed references to be provided to applicants during examination: OP 3308-003

Learning Objective: MC-05095 Describe the process for verifying the position of the following types of equipment: A) Manual Valves B) Throttle Valves C) Locked Valves. (As available)

Question Source: New

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
Knowledge of the process for managing shutdown	Group #	2	2
maintenance activities, such as risk assessments,	K/A #	GEN.2.2.18	
work prioritization, etc.	Importance Rating	2.6	3.9

Proposed Question:
Current Plant Conditions:

- The plant is in MODE Zero.
- An "A" Train electrical outage is in progress.
- The "B" Spent Fuel Pool Cooling Pump is running.

What is an additional "Defense-In-Depth" requirement for Decay Heat Removal that exists specific to this condition?

A. An electrician, carrying a beeper, must be available to establish temporary power to the "A" Spent Fuel Pool Cooling Pump.

B. An electrician, carrying a beeper, must be available to establish temporary power to the "A" RPCCW Pump.

C. The SBO Diesel must be available to power the "B" Train, and the time to reach boiling in the Spent Fuel Pool must be greater than 30 minutes.

D. Spent Fuel Pool level must be maintained at greater than 23 feet, with the Fire Protection Water System available as a backup water source for the Fuel Pool.

Proposed Answer: A

Explanation (Optional):
To maintain defense in depth for Spent Fuel Pool Cooling during a full core offload, the following is required:

1. One protected Train Spent Fuel Cooling (SFC) Pump.
2. An electrician, carrying a beeper, must be available to establish temporary power to the "A" SFC Pump ("A" correct).
3. Either one SFC Heat Exchanger with two RPCCW pumps aligned to the associated RPCCW Train and two SWP Pumps on the associated Train; or, two SFC Heat Exchangers each with one SWP Pump and RPCCW Pump.

"B", "C", and "D" are wrong, since they are not specific requirements on this list. "B" is plausible, since backup RPCCW is required, but the swing pump is aligned to the associated train. "C" is plausible, since backup power is required (but via temporary jumper), and the SBO cannot normally be credited for Shutdown Risk unless time to core boil is >30 minutes, due to the time it takes to manually start and load the SBO diesel. "D" is plausible, since the Spent Fuel Pool must normally be maintained above 23 feet to credit it for Shutdown Risk, and Fire Protection Water is a backup water source for the Spent Fuel Pool.

Technical Reference(s): OP 3260A (Rev,016-07), Section 1.4.9
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05364 Discuss the concept of "Defense-In-Depth" as it applies to Spent Fuel Pool Cooling. (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 71	Tier #	3	3
Ability to obtain and interpret electrical and	Group #	2	2
mechanical drawings	K/A #	GEN.2.2.41	
	Importance Rating	3.5	3.9

Proposed Question:

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

1. The BOP operator reports that the RED, GREEN, and AMBER lights for the 4KV Feeder breaker to Load Center 32T on Main Board 8 are NOT lit.
2. The operator does a lamp check and the RED, AMBER and GREEN lights DO NOT light.
3. A PEO goes to the breaker and reports the white auxiliary circuit light is the only light lit.

Using ESK-5A, attached to this exam, what is the status of the breaker, and what is a possible cause?

A. The breaker is closed, and indication is lost due to an open circuit in the TRIP COIL.

B. The breaker is open, and indication is lost due to an open circuit in the TRIP COIL.

C. The breaker is closed, and indication lost due to a blown UC fuse.

D. The breaker is open, and indication lost due to a blown UT fuse.

Proposed Answer: A

Explanation (Optional): The trip coil is in series with the local and remote RED lights for the breaker. The GREEN light is not in the path for the Trip Coil. If the Trip coil had an open, the red lights would go out. The breaker would not open because the Trip Coil is energize-to-actuate ("B" wrong). So the breaker remains closed (the green lights and amber lights will not come on). Going to test routes the MB red light through the same path (as the trip coil) as the green and amber lights, so they will not illuminate ("A" correct). The UC and UT fuses are OK, as evidenced by the white light ("C" and "D" wrong). "B", "C", and "D" are plausible, since each of these failures will create indication problems for the breaker, and certain circuit failures would cause the breaker to open.

Technical Reference(s): ESK 5A (Rev. 13)
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: ESK 5A

Learning Objective: MC-04130 Describe the function and electrical operation of the following breaker control circuit components: A. 52x Coil and Circuit B. 52y Coil and circuit. C. Trip Coil and circuit D. UC Fuses and circuit E. UT Fuses and circuit. (As available)

Question Source: Bank #68182

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 # 72	Tier #	3	3
Knowledge of radiological safety procedures pertaining to licensed operators such as radiation monitor alarms, Ctmt entry etc	Group #	3	3
	K/A #	GEN.2.3.13	
	Importance Rating	3.4	3.8

Proposed Question:

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

1. Control Room Ventilation Makeup Air Supply Radiation Monitors 3HVC16A and B both go into HI ALARM.
2. The crew enters AOP 3573 *Radiation Monitor Alarm Response*.
3. A PEO is dispatched to close and dog the Control Building pressure boundary doors.

In accordance with AOP 3573, what other actions will the crew take in response to the high radiation condition?

A. Verify Outside Air Isolation Valves (3HVC*AOV25 and 26) are <u>open</u>; and after one minute, verify Air Bottle Isolation Valves 3HVC*SOV74A/B open automatically.

B. Verify Outside Air Isolation Valves (3HVC*AOV25 and 26) are <u>closed</u>; and after one minute, verify Air Bottle Isolation Valves 3HVC*SOV74A/B open automatically.

C. Stop the kitchen exhaust fan, verify the "A" Control Room Filter Fan (3HVC*FN1A) automatically started; and verify Outside Air Isolation Valves (3HVC*AOV25 and 26) are <u>open</u>.

D. Stop the kitchen exhaust fan, verify the "A" Control Room Filter Fan (3HVC*FN1A) automatically started; and verify Outside Air Isolation Valves (3HVC*AOV25 and 26) are <u>closed</u>.

Proposed Answer: C

Explanation (Optional): A CBI automatically aligns the control building ventilation system to the filtered, pressurized recirculation mode upon receipt of a CBI signal. Therefore, operators will stop the kitchen exhaust fan, since its outlet damper has automatically closed, verify the "A" Control Room Filter Fan (3HVC*FN1A) automatically started; and verify Outside Air Isolation Valves (3HVC*AOV25 and 26) are <u>open</u> ("C" correct and "D" wrong). "A" and "B" are wrong, since the crew will not verify Air Bottle Isolation Valves 3HVC*SOV74A/B open. "A" and "B" are plausible, since verifying Air Bottle Isolation Valves 3HVC*SOV74A/B open automatically after one minute was a required action prior to the CBI modification, made in the late 2008. "D" is plausible, since AOV25 and 26 used to automatically close on a CBI, and the filters can be run on full recirculation with these valves closed (but they are kept open to maintain the control room at a positive pressure).

Technical Reference(s): AOP 3573 (Rev. 018), Attachment A, page 5 of 12
(Attach if not previously provided) OP 3314F (Rev. 022), section 4.13.2
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06051 Describe operation of HVC System under the following… High Radiation detected by HVC*RE16A or B… (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10, 41.11, and 41.12

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 73	Tier #	3	3
Knowledge of radiation hazards that may arise during activities	Group #	3	3
	K/A #	GEN.2.3.14	
	Importance Rating	3.4	3.8

Proposed Question:

With the plant at 100% power, area radiation monitor RMS16-1 (VCT and Boric Acid Tank area) goes into HI ALARM. The RO reviews the rough log and notes the following evolutions have recently been conducted:

- A Liquid Waste Discharge was commenced.
- The Degassifier was shutdown.
- The Boron Evaporator was started up.
- A Solid Waste System resin transfer was commenced.

Which of the above activities was the likely cause of the alarm?

A. The Liquid Waste discharge.

B. The shutdown of the Degassifier.

C. The startup of the Boron Evaporator.

D. The resin transfer.

Proposed Answer: B

Explanation (Optional): The degassifier degasses the letdown stream prior to entry into the VCT. With the degassifier shutdown, a RMS16-1 alarm can be anticipated as radioactive gasses accumulate in the VCT ("B" correct). "A", "C", and "D" are plausible, since they involve movement of radioactive material through the auxiliary building.

Technical Reference(s): AOP 3573 (Rev. 018), Attachment B, page 3
(Attach if not previously provided)
(including version/revision number)

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, and the basis for each. (As available)

Question Source: Bank #76167

Question History: Millstone 3 2002 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.11, 43.4 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 74	Tier #	3	3
Knowledge of the emergency plan	Group #	4	4
	K/A #	GEN.2.4.29	
	Importance Rating	3.1	4.4

Proposed Question:

A MP3 Control Room Operator is attending his training week, and is currently performing In-Plant JPMs during normal working hours.

An actual ALERT-Charlie 1 classification is declared for Millstone 3.

To what location is the CO required to report?

A. The EOF

B. The OSC Assembly Area (Bldg 475 Cafeteria)

C. The Tech Support Center/Ops Support Center

D. The Control Room

Proposed Answer: B

Explanation (Optional): An RO is labeled as an "On Shift" position. If the ALERT is declared while on duty, the RO reports to the Control Room ("D" plausible). If the ALERT is declared while off duty during normal business hours, the RO reports to the OSC assembly area, which is the Bldg 475 Cafeteria ("B" correct, "A", "C", and "D" wrong). Other SERO members responding from offsite during normal business hours, report to the designated response facility [TSC ("C" plausible), EOF ("A" plausible), etc.].

Technical Reference(s): MP-26-EPA-FAP01 (Rev. 003), Sections 2.9, 2.10, and Att. 4, page 4 of 7.
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-01295 Identify the designated assembly areas for on duty, as well as off-duty SERO personnel if assembly has been ordered by CRDSEO. (As available)

Question Source: New

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
Ability to prioritize and interpret the significance of annunciators	Group #	4	4
	K/A #	GEN.2.4.45	
Proposed Question:	Importance Rating	4.1	4.3

With the plant operating at 100% power, an earthquake occurs, and the following annunciators are received.

- EARTHQUAKE (MB1)
- RPCCW SURGE TK LEVEL LO (MB1)
- TDFW PP A SUCTION PRESSURE LO (MB5)
- GEN SEAL OIL PUMP DIS PRES LO (MB7)
- GEN SEAL OIL TO H2 DP LO (MB7)
- CRDM CLG FAN A AUTO TRIP (VP1A)

The board operators make the following reports:

- RPCCW Surge Tank level is 90% and decreasing slowly.
- Feed Pump suction pressure is 260 psig and stable.
- Generator hydrogen pressure is 28 pounds and decreasing slowly.
- The "A" CRDM Cooling Fan amber light is lit.
- CRDM Shroud Inlet Temperature is 97°F and stable.

Which of the below listed annunciators is the highest priority for the crew?

A. RPCCW SURGE TK LEVEL LO.

B. TDFW PP A SUCTION PRESSURE LO.

C. GEN SEAL OIL TO H2 DP LO.

D. CRDM CLG FAN A AUTO TRIP.

Proposed Answer: C

Explanation (Optional): Seal oil low DP with hydrogen pressure <30 psig requires a trip. An annunciator requiring a reactor trip is a higher priority ("C" correct) than one requiring an AOP entry ("A" wrong), a downpower ("B" wrong), or an orderly plant shutdown ("D" wrong). "A", "B", and "D" are plausible since these annunciators require prompt action.

Technical Reference(s): OP 3353.MB1C (Rev. 005-13), 2-7B, OP 3353.MB5A (Rev. 004), 3-6
(Attach if not previously provided) OP 3353.MB7A (Rev. 003-03), 1-5
(including version/revision number) OP 3353.VP1A (Rev. 003-00), 4-7

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04703 Given a plant condition or equipment malfunction relating to the GMO system, determine when the turbine is required to be tripped, or when the generator must be shutdown. (As available)

Question Source: Modified Bank #78788 Parent Question Attached

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.10 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 76	Tier #		1
RCP Malfunctions:	Group #		1
Determine/interpret the cause of an RCP failure	K/A #	APE.015/017.AA2.01	
	Importance Rating		3.5

Proposed Question:

With the plant at 100% power, the following three annunciators are received:

- RCP HI RANGE LKG FLOW HI (MB3B, 2-10)
- RCP A NO. 2 SEAL LEAKOFF HI (MB4B, 1-2)
- RCP A STNPIPE LEVEL HI (MB4B, 2-2A)

After the initial transient, the RO reports RCP A seal parameters have stabilized at the following values:

- RCP A #1 Seal leakoff flow: 7.1 gpm.
- RCP A Seal Inlet Temperature: 145°F and increasing.
- RCP A #2 Seal leak rate (based on CDTT Level Trend): 3.0 gpm.

What is the status of the "A" RCP seal package, and what action is the crew required to take?

A. The #1 seal is failing only. The crew is required to commence an orderly plant shutdown.

B. The #2 seal is failing only. The crew is required to trip the reactor, stop the "A" RCP, and go to E-0.

C. The #1 and #2 seals are failing. The crew is required to commence an orderly plant shutdown.

D. The #1 and #2 seals are failing. The crew is required to trip the reactor, stop the "A" RCP, and go to E-0.

Proposed Answer: D

Explanation (Optional): The #2 seal has failed, since #2 Seal leakoff has increased ("A" wrong, "B" plausible). The #1 seal is also failing, since its leakoff flow has also increased ("B" wrong, and "A" plausible). If the #1 seal was functioning properly with a failed #2 seal, its leakoff flow would decrease as more flow passes through the failed #2 seal. With the #2 seal failed, water will flow past the #3 seal and up into the standpipe, causing a standpipe high level. "D" is correct and "C" wrong, since, even though the #1 and #2 seal leak rates don't individually require a reactor trip ("C" plausible), the combined leakage (>8 gpm) does.

Technical Reference(s): OP 3353.MB4B, 1-2 (Rev. 004-09)
(Attach if not previously provided) OP 3353.MB4B, 2-2A (Rev. 004-09)
(including version/revision number) OP 3353.MB3B, 2-10 (Rev. 006-10)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05434 Explain the effects of, and describe the required actions for the following RCP seal failures: A. #1 Seal Failure, B. #2 Seal Failure, C. #3 Seal Failure (As available)

Question Source: New

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
Loss of RHR System:	Group #		1
Determine/interpret existence of	K/A #	APE.025.A2.06	
proper RHR overpressure protection	Importance Rating		3.4

Proposed Question:

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

1. The "A" PORV block valve is closed due to a leaky PORV.
2. When RCS hot leg temperature reached 340°F, the "A" train of RHR was placed in service in the cooldown mode.
3. When RCS hot leg temperature reached 245°F, the "B" train of RHR was also placed in service in the cooldown mode.

Current conditions:

- The "B" train of RHR has just been isolated from the RCS due to a significant RHR piping leak.
- Hot leg temperatures are 220°F.
- The "B" RCP is running.

What is the status of Cold Overpressure Protection, and does LCO 3.4.9.3 "Overpressure Protection Systems" need to be entered?

A. Adequate Cold Overpressure Protection <u>is</u> available, since Cold Overpressure Protection is not required until MODE 5. LCO 3.4.9.3 does NOT need to be entered.

B. Adequate Cold Overpressure Protection <u>is</u> available, since the crew has already armed COPPS, and a single PORV and one RHR Suction Relief are still available. LCO 3.4.9.3 does NOT need to be entered.

C. Adequate Cold Overpressure Protection is <u>not</u> available, since the crew has not yet armed COPPS, and was relying on both RHR suction relief valves. The crew must enter LCO 3.4.9.3.

D. Adequate Cold Overpressure Protection is <u>not</u> available, since either 2 PORVs or two RHR Suction Relief Valves must be available to satisfy Cold Overpressure Protection requirements. The crew must enter LCO 3.4.9.3.

Proposed Answer: B

Explanation (Optional): This question requires detailed knowledge of GOP plant conditions and of LCO 3.4.9.3. COPPs is required ≤226°F ("A" wrong), from either 2 PORVs, 2 RHR suction relief valves, or one of each ("D" wrong). "B" is correct, and "C" wrong, since COPPS is armed by procedure when hot leg temperatures reach 250°F. "A" is plausible, since COPPS is only required at cold temperatures. "C" is plausible, since both trains of RHR are normally available in the cooldown mode during initial plant cooldown to MODE 5, and two RHR suction reliefs provides adequate COPPS protection. "D" is plausible, since normally, two RHR suction relief valves or two PORVs are available for COPPS.

Technical Reference(s): OP 3208 (Rev. 021-03), steps 4.3.4, Note prior to step 4.3.5, and 4.3.31
(Attach if not previously provided) OP 3310A (Rev. 016-12), step 4.5
(including version/revision number) Tech Spec LCO 3.4.9.3 (Amendment 197)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05459 Given a failure, partial or complete, of the RHR system, describe the effects on the system and on interrelated systems. (As available)

Question Source: New

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 # 78	Tier #		1
Steamline Rupture:	Group #		1
Knowledge of alarms, indications, or annunciator	K/A #	APE.040.GEN.2.4.31	
response procedures	Importance Rating		4.1

Proposed Question:
With the plant initially at 100% power, a turbine runback occurs, and the following sequence of events occurs:

1. Safety Valves lift on all 4 SGs.
2. A steamline break occurs in the Main Steam Valve Building.
3. SIS and MSI actuate.
4. The crew successfully carries out all applicable steps in E-0 *Reactor Trip or Safety Injection*.
5. The crew transitions to E-2 *Faulted Steam Generator Isolation*.

While in E-2, the BOP operator reports the following indications exist:

- Several SG Safety Valves still indicate OPEN.
- The following valve position indicators no longer indicate on MB5:
  - 3FWS*LV560 (“B” SG Feed Reg Bypass Valve)
  - 3FWS*FCV530 (“C” SG Feed Reg Valve)
  - 3MSS*CTV27A (“A” SG Main Steam Isolation Valve)

What action is the US required to take in response to these indications?

A. The crew is required to consult with the Assistant Director of Technical Support (ADTS) to determine if the Safety Valves should be gagged closed, since they are still open. The crew is required to remove power to the other valves, since this will ensure that they fail closed.

B. The crew is required to consult with the Assistant Director of Technical Support (ADTS) to determine if the Safety Valves should be gagged closed, since they are still open. It is acceptable to rely on the previously reported “closed” indications for the other valves, since no automatic signals would open the valves once closed.

C. Backup indications, including local observations, should be used to determine Safety Valve position, since these temperature switches may be indicating erroneously. The crew is required to remove power to the other valves, since this will ensure that they fail closed.

D. Backup indications, including local observations, should be used to determine Safety Valve position, since these temperature switches may be indicating erroneously. It is acceptable to rely on the previously reported “closed” indications for the other valves, since no automatic signals would open the valves once closed.

Proposed Answer: D

Explanation (Optional): “D” is correct, since the safety valve flow indications are derived from differential temperature switches, and they may erroneously indicate flow if the common drain header is warmed by previous steam releases. Flow indication should be verified by local observation and other plant responses (“A” and “B” wrong). Also, an unisolable steamline break in the MSVB will disable valve position indication for valves in the MSVB after several minutes. If valve position indication is no longer available, previous indication of valve closure is acceptable (“C” wrong). Alternatively, the valves may be ensured closed by removing power since the valves are failed closed (“A” and “C” are plausible). “B” is plausible, for most MB indications, the crew is not required to verify conditions locally prior to “trusting” the indication.

Technical Reference(s): OP 3272 (Rev. 008-08), Attachment 3, Sheet 5 of 12.
(Attach if not previously provided) E-2 (Rev. 011), Caution prior to step 5.
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-0 Given a set of plant conditions, determine the required actions to be taken in accordance with E-2 Faulted Steam Generator Isolation. (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:		Level	RO	SRO
Question # 79		Tier #		1
Loss of Nuclear Service Water:		Group #		1
Determine/interpret normal values for SWS header flow		K/A #	APE.062.AA2.05	
rate and flow rates to components		Importance Rating		2.5

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

1. The RPCCW HX SW FLOW HI/LO (MB1C, 1-1A) annunciator is received.
2. The RO reports the following Service Water flow indications on MB1:
   - 3SWP-F43A, "SERVICE WTR DIS FLOW" "RPCCW HX:" 11,800 gpm
   - 3SWP-F43B, "SERVICE WTR DIS FLOW" "RPCCW HX:" 8,800 gpm

Which train of Service Water has the flow problem; and what action is the crew required to take?

A. A pipe break exists on the "A" Train. The US will enter AOP 3560 *Loss of Service Water*, since AOP 3560 will provide direction to isolate the break.

B. A pipe break exists on the "A" Train. The US will enter the ARP to isolate the break by closing Service Water Supply Valve 3SWP*MOV50A, and then go to AOP 3560 *Loss of Service Water*.

C. Flow blockage exists on the "B" Train. The US will enter AOP 3560 *Loss of Service Water*, since AOP 3560 will direct the crew to place the standby RPCCW Heat Exchanger in service.

D. Flow blockage exists on the "B" Train. The US will enter the ARP to check SWP to RPCCW Valves aligned properly, check for RPCCW Heat Exchanger fouling; and then go to AOP 3560, *Loss of Service Water*.

Proposed Answer: B
Explanation (Optional): Normal SWP Flowrate to an RPCCW HX is about 8,800 gpm, so Train "A" is showing excessive flow, indicating a potential pipe break, while train "B" is showing a normal flowrate ("C" and "D" wrong). "C" and "D" are plausible, since flow is lower in the "B" train than the "A" train, AOP 3560 takes actions for a loss of service water, and the ARP directs the actions in "D" if flow is low. "B" is correct, since a leak downstream of 3SWP*MOV50A is isolable, and will be isolated by the ARP. "A" is wrong, since AOP 3560 does not address the initial required actions for a pipe break, since the initial actions are provided in the ARPs. "A" is plausible, since AOP 3560 takes actions for a loss of service water.

Technical Reference(s):	OP 3353.MB1C (Rev. 005-13), 1-1A, entry conditions, and step 8.2
(Attach if not previously provided)	AOP 3560 (Rev. 008), Note prior to step 1.
(including version/revision number)	FSAR (Rev. 21.3) Table 9.2-1

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05719 Given a failure, partial or complete, of the Service Water System, determine the effects on the system, and on interrelated systems. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 80	Tier #		1
Loss of Instrument Air:	Group #		1
Ability to explain and apply system limits and precautions	K/A #	APE.065.GEN.2.1.32	
	Importance Rating		4.0

Proposed Question:

Initial Conditions:

- The plant is at 100% power.
- The Instrument Air System Emergency Air Dryer is in service, due to a failed normal air dryer heater.
- A cold front has arrived, resulting in the following:
    - Outside air temperature has dropped to 30°F.
    - Wave height is elevated at the intake structure.

The following sequence of events occurs:

1200: Instrument air pressure starts decreasing at a moderate rate.
1203: The crew enters AOP 3562 *Loss of Instrument Air*.
1212: The RO reports letdown has isolated.
1214: The BOP reports he cannot maintain SG level with Feed Regulating Valve demand at 100%.

In accordance with AOP 3562, which plant condition first required the US to direct a reactor trip; and what specific action based on the status of instrument air is the US required to ensure the crew carries out?

A. A reactor trip was first required when Letdown isolated. The US is required to direct the crew to place traveling screens in SLOW-1, and increase surveillance of the Intake Structure, since traveling screen DP indication and the Main Circulating Pump screen DP trip relays may not function.

B. A reactor trip was first required when Letdown isolated. The US is required to dispatch a PEO to blowdown the emergency air dryer via a hose to the nearest floor drain, since excessive moisture in the instrument air lines while on the emergency air dryer may have caused the loss of air pressure.

C. A reactor trip was first required when Feedwater control was lost. The US is required to direct the crew to place traveling screens in SLOW-1, and increase surveillance of the Intake Structure, since traveling screen DP indication and the Main Circulating Pump screen DP trip relays may not function.

D. A reactor trip was first required when Feedwater control was lost. The US is required to dispatch a PEO to blowdown the emergency air dryer via a hose to the nearest floor drain, since excessive moisture in the instrument air lines while on the emergency air dryer may have caused the loss of air pressure.

Proposed Answer: C

Explanation (Optional): A reactor trip is required either when instrument air pressure starts decreasing at a rapid rate, or when feed control is lost ("A" and "B" wrong). "C" is correct, since screen DP indicators/circ pump trip relays are pneumatic, and will not function on a loss of air pressure. This is especially important with increased wave action. "D" is wrong, since the emergency air dryer blowdown liquid contains sodium/magnesium chlorite, which is not permitted to be discharged to floor drains. It should be collected in a non-metallic container. "A" and "B" are plausible, since letdown isolates on a loss of air. "B" and "D" are plausible, since an operational concern exists when on the emergency air dryer during cold weather. Moisture may collect, and subsequently freeze during cold weather, and outside air temperature is below freezing.

Technical Reference(s): AOP 3562 (Rev. 006), steps 1 and 10, including associated cautions
(Attach if not previously provided) OP 3332A (Rev. 015-02), notes associated with steps 4.3.1 and 4.3.2
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05322 Describe major administrative or procedural precautions and limitations placed on operation of plant air systems, including the basis for each. (As available)

Question Source: New

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 # 81	Tier #		1
Inadequate Heat Transfer – Loss of Secondary Heat Sink:	Group #		1
Determine / interpret adherence to appropriate procedures	K/A #	EPE.W/E05.EA2.2	
	Importance Rating		4.3

Proposed Question:

Current Conditions:

- The crew is in FR-H.1 *Response to Loss of Secondary Heat Sink.*
- The crew has established RCS "bleed and feed".
- Both Motor Driven AFW pumps are now available for service.
- A flowpath has been established to allow feeding the intact SGs.
- CTMT temperature is 130°F.
- Core Exit Thermocouples are increasing.
- SG conditions are as follows:
  - "A" SG is ruptured. Narrow Range level is 6%.
  - "B" SG is faulted and completely depressurized.
  - "C" SG is intact. Wide Range level is 8%.
  - "D" SG is intact. Wide Range level is 8%.

How is the crew required to establish AFW flow?

A. Establish 100 gpm feed flow to "C" **and** "D" Steam Generators.

B. Establish 100 gpm feed flow to "C" **or** "D" Steam Generator.

C. Establish maximum feed flow to "C" **and** "D" Steam Generators.

D. Establish maximum feed flow to "C" **or** "D" Steam Generator.

Proposed Answer: D

Explanation (Optional): With a feed source available and all intact SGs < 12% WR, and Core Exit TCs INCREASING, the crew is required to establish maximum feed flow rate ("A" and "C" wrong) to only one SG ("D" correct, "B" wrong). "A", "B", and "C" are plausible since each of these feed rates could be required based on SG levels and CETC temperature trends, and based on whether Bleed and Feed has been initiated yet.

Technical Reference(s): FR-H.1 (Rev. 020), steps 3 and 19
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07461 (SRO, STA) Given a set of plant conditions, determine the required actions to be taken per FR-H.1. (As available)

Question Source: Bank #63912

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 82	Tier #		1
Accidental Liquid Waste Release:	Group #		2
Ability to perform specific system and integrated plant procedures during all modes of plant operation	K/A #	APE.059.GEN.2.1.23	
	Importance Rating		4.4

Proposed Question:

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

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

A. A temporary monitor must be used with its alarm setpoint set more conservatively than the LWS70 setpoint to allow the operator sufficient time to manually stop the discharge in the event an alarm condition occurs.

B. Best efforts must be made to repair the instrument; and, at least two independent samples, independent release calculations, and independent discharge valve lineups must be performed prior to initiating the discharge.

C. Best efforts must be made to repair the instrument; the "A" LLWDT must be recirculated an additional 15 minutes, and independent discharge valve lineups must be performed prior to initiating discharge.

D. Samples must be taken every 15 minutes while the discharge is in progress, to verify the effluent is within Technical Specifications limits.

Proposed Answer: B

Explanation (Optional): "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 (Rev. 026-00) (REMODCM) Section V.C.1, page 142
(Attach if not previously provided)	MP-22-REC-BAP01 (Rev. 026-00) Table V.C.-1, page 143
(including version/revision number)	MP-22-REC-BAP01 (Rev. 026-00) Table V.C.-1 Action Statements, page 145

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05931 Given a plant condition requiring the use of AOP 3573, identify applicable Technical Specification and/or REMODCM Requirements. (As available)

Question Source: Bank #74490

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 10CFR55.41.10, 43.2, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 83	Tier #		1
High Containment Radiation:	Group #		2
Knowledge of EOP mitigation strategies	K/A #	EPE.W/E16.GEN.2.4.6	
	Importance Rating		4.7

Proposed Question:

The reactor has tripped, and the crew is progressing through the EOP network when the following sequence of events occurs:

1. The crew enters EOP 35 FR-Z.3 *Response To High Containment Radiation Level.*
2. The STA reports CTMT Dew Point is normal.
3. Sample results show that both particulate and I-131 levels are elevated in Containment atmosphere.
4. The US requests ADTS concurrence on starting one Containment Air Filtration (CAF) Fan.

How effective will the CAF fan be in reducing radiation levels; and, per FR-Z.3, which other equipment will specifically be considered for use in lowering CTMT radiation levels?

A. The CAF System will be effective in reducing both particulate <u>and</u> I-131 levels in CTMT. The CTMT Spray Pumps will also be considered for use.

B. The CAF System will be effective in reducing both particulate <u>and</u> I-131 levels in CTMT. The CTMT Purge System will also be considered for use.

C. The CAF System will be effective in reducing particulates but <u>not</u> I-131 levels in CTMT. The CTMT Spray Pumps will also be considered for use.

D. The CAF System will be effective in reducing particulates but <u>not</u> I-131 levels in CTMT. The CTMT Purge System will also be considered for use.

Proposed Answer: A

Explanation (Optional): FR-Z.3 samples CTMT atmosphere, considers the use of CTMT Air Filtration, and considers use of CTMT Spray System ("B" and "D" wrong). "B" and "D" are plausible, since the CTMT Purge System would remove activity from CTMT, and is used in the EOP network as a backup hydrogen removal path from CTMT during an accident. The CAF System will remove both particulates with its HEPA filters and I-131 with its charcoal bed adsorbers ("A" correct and "C" wrong). "C" is plausible, since mechanical filters are not effective in removing gaseous activity.

Technical Reference(s): FR-Z.3 (Rev. 5)
(Attach if not previously provided) FSAR (Rev. 21.3) Section 9.4.7.1
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major action categories within EOP 35 FR-Z.3 (As available)

Question Source: Bank #74362

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 # 84	Tier #		1
RCS Overcooling – PTS: Knowledge of events	Group #		2
that must be reported to internal organizations	K/A #	EPE.W/E08.GEN.2.4.30	
or eternal agencies, such as the State or NRC	Importance Rating		4.1

Proposed Question:

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

<u>Time:</u> <u>Event:</u>

0000: A steam leak occurs inside CTMT.
0105: The crew commences a Plant Shutdown as required by LCO 3.6.1.4 CONTAINMENT PRESSURE.
0340: The steam break gets worse, and the crew manually trips the reactor.
0405: Safety Injection automatically actuates.
0510: An RCS Integrity Red Path is received on SPDS.

Which was the <u>first</u> event that required an NRC notification to be made?

A. The initiation of the plant shutdown

B. The manual actuation of the Reactor Protection System

C. The ECCS discharge into the Reactor Coolant System

D. The RCS Integrity Red Path on SPDS

Proposed Answer: A.

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since the initiation of a plant shutdown required by Tech Specs requires a 4 hour report per 10CFR50.72(b)(2)(i), and this event occurred first in the timeline. "B" and "C" are plausible, since these events also require a 4 hour report. "D" is plausible, since this meets the threshold for an ALERT emergency classification due to the potential loss of the RCS barrier.

Technical Reference(s): RAC 14 (Rev. 002-06), Attachment 1, Sheet 1 of 4.
(Attach if not previously provided) MP-26-FAP06-003 (Rev. 005) EAL Tables, Barrier Failure Reference Table
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-00016 (SRO) Given a plant condition or equipment malfunction, use provided reference material to determine… required federal and/or state reporting requirements… (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 85	Tier #		1
Loss of All AC Recovery with the SBO Diesel: Ability to interpret	Group #		2
control room indications to verify status and operation of a system,	K/A #	Site spec.GEN.2.2.44	
and understand how operator actions affect plant/system conditions	Importance Rating		4.4

Proposed Question:

The following sequence of events occurs:

1. The plant trips due to a loss of all AC power.
2. The crew enters ECA-0.0 *Loss of All AC Power*.
3. The crew restores power to Bus 34C from the SBO diesel.
4. The crew transitions to ECA-0.3 *Loss of All AC Power -- Recovery with the SBO Diesel.*
5. The crew starts the "A" Charging Pump per ECA-0.3, step 6.

Current Conditions are as follows:

- Pressurizer level: 10%
- RCS pressure: 1550 psia
- Core Exit Thermocouples: 564°F
- CTMT temperature: 125°F

In accordance with ECA-0.3, what action is the crew required to take?

A. Open Charging Flow Control Valve 3CHS*FCV 121 to increase PZR level above 16%.

B. Open one charging pump cold leg injection valve and increase PZR level above 16%.

C. Actuate Safety Injection, and remain in ECA-0.3 *Loss of All AC Power -- Recovery with the SBO Diesel.*

D. Actuate Safety Injection, and transition to ECA-0.2 *Loss of All AC Power Recovery with SI required.*

Proposed Answer: B

Explanation (Optional): The crew has just started a Charging Pump, so the crew has just completed step 6. With the SBO diesel as the only source of power, significant loading limitations exist, so the crew will not transition to another EOP, since other EOPs assume at least one emergency bus is available ("D" wrong). The crew will not actuate SIS ("C" wrong), since SI is directed to be reset to allow manual loading of equipment (and avoid overloading the SBO diesel) per the caution prior to step 1 of ECA-0.3. "B" is correct, and "A" wrong, since the cold leg injection valve will supply the maximum amount of water from one charging pump with low Pzr level. "A" is plausible, since this would raise PZR level, the Charging pump is currently supplying water through FCV 121, and this action is directed in other procedures, such as AOP 3555 RCS leak. "C" and "D" are plausible, since actuating SI would raise PZR level, PZR level is below the SI reinitiation setpoint on the foldout page of several EOPs. Also, ECA-0.2 would be the correct choice if offsite power or an EDG were supplying power.

Technical Reference(s): ECA-0.3 (Rev. 013) Caution prior to step 1, and Steps 6 and 7
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07411 Given a set of plant conditions, determine the required actions to be taken per ECA-0.3. (As available)

Question Source: Bank #67595

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 # 86	Tier #		2
Reactor Protection:	Group #		1
Ability to evaluate the plant and make	K/A #	012.GEN.2.1.7	
operational judgments	Importance Rating		4.7

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

1. The CONTAINMENT PRESSURE HI-1 Annunciator comes in on MB2.
2. The RO reports all CTMT pressure instruments indicate 13.7 psia.
3. The RO reports that only one CTMT PRESS HI-1 Bistable is lit on MB2.
4. The crew enters the appropriate procedures.
5. The crew is preparing to place the Train A SSPS Multiplexer Test switch in the "A+B" position to assist in distinguishing whether the failure is within SSPS or at the protection channel.
6. The extra operator reports that the red GENERAL WARNING lamp on 3RPS*RAKLOGB is lit.
7. The extra operator also reports that the Train B SSPS "Multiplexer Test" switch is in NORMAL at 3RPS*RAKLOGB.

Which procedure provides direction on distinguishing whether the failure is within SSPS or at the protection channel; and is the crew required to place the Train A SSPS Multiplexer Test switch in the "A+B" position?

A. The CONTAINMENT PRESSURE HI-1 ARP provides guidance. The crew will NOT select "A+B", since this is only required if an associated instrument has also failed.

B. The CONTAINMENT PRESSURE HI-1 ARP provides guidance. The crew WILL select "A+B", since if this causes the affected bistable light to start flashing, the RPS Bistable can be tripped without further troubleshooting of SSPS.

C. AOP 3571 *Instrument Failure Response* provides guidance. The crew will NOT select "A+B", since this step will result in a reactor trip with the plant in this configuration.

D. AOP 3571 *Instrument Failure Response* provides guidance. The crew WILL select "A+B", since if this causes the affected bistable light to start flashing, I&C will need to troubleshoot SSPS prior to tripping the RPS Bistable.

Proposed Answer: C

Explanation (Optional): "A" and "B" are wrong, since the CTMT Pressure HI-1 ARP does not adequately address troubleshooting a single failed bistable. "A" is plausible, since ARP entry conditions are met; and no instrument has failed, making this an unusual entry into AOP 3571. "B" is plausible, since this action would be correct if the opposite train GENERAL WARNING light was not illuminated and the bistable light did not start flashing. "C" is correct, and "D" is wrong, since with a GENERAL WARNING on train B, taking this switch to "A + B" will mean both trains of SSPS are not in a normal lineup, and a reactor trip will occur. "D" is wrong, but plausible, since this would be the correct answer if the opposite train GENERAL WARNING light was not illuminated.

Technical Reference(s): AOP 3571 (Rev. 009-04), Attachment R, step 1.d.
(Attach if not previously provided) OP 3353.MB2A (Rev. 003-01), 5-3
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03976 Describe the major action categories contained within AOP 3571 (As available)

Question Source: Bank 80913

Question History: Last NRC Exam (Millstone 3 2007 NRC Exam)

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 87	Tier #	2	2
Containment Spray: Knowledge of low power/ shutdown	Group #	1	1
implications in accident (LOCA) mitigation strategy	K/A #	026.GEN.2.4.9	
Proposed Question:	Importance Rating		4.2

Initial Conditions:

- A plant cooldown is in progress per OP 3208 *Plant Cooldown*.
- The "A" RHR Pump is running in the "Cooldown" Mode.
- RCS temperature is 240°F.

The following sequence of events occurs:

1. A large break LOCA occurs.
2. Containment pressure reaches 25 psia.
3. All signals actuate as designed for current plant conditions.
4. The crew chooses to enter E-0 *Reactor Trip and Safety Injection*, and perform a step by step evaluation to determine if specified actions are still applicable for current plant conditions.

How will current plant conditions impact the crew's use of CTMT Spray while progressing though the EOP network?

A. Quench Spray and RSS pumps are not readily available for use, since their breakers have been racked down.

B. Quench Spray and RSS pumps will have to be manually started from the Main Boards, since their switches are in Pull-To-Lock; and the discharge valves will have to be manually opened.

C. The crew will need to manually actuate CDA from Main Board 2, since the automatic CDA signal has previously been blocked.

D. Quench Spray and RSS steps will be carried out as written, since the QSS and RSS systems will respond to the CDA in the same manner as they would if the LOCA had occurred in MODE 1.

Proposed Answer: D

Explanation (Optional): This question requires detailed knowledge of Technical Specification requirements in lower MODES of operation. RCS temperature is below the point where ECCS is blocked, COPPS is placed in service, and RHR is placed in service in the cooldown mode. However, CTMT Spray is required to remain OPERABLE throughout MODE 4 ("D" is correct, and "A", "B", and "C" wrong). "A" is plausible, since, after MODE 5 is entered, QSS and RSS will be placed in Pull-To-Lock, and their breakers can be racked down. "B" is plausible, since the ECCS Pumps are placed in Pull-To-Lock when RCS temperature is below 340°F to protect against a mass addition event. "C" is plausible, since, after a cooldown is commenced, automatic SIS is blocked from MB2. Also, after MODE 5 is entered for a long-term outage, the crew can block the CDA signal to RSS Pumps and Valves.

Technical Reference(s): OP 3208 (Rev. 021-03), steps 4.3.3, 4.3.5, 4.3.31, 4.3.40, and 4.3.42.
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07503 (SRO) Given a set of plant conditions, determine the required actions to be taken per OP 3208. (As available)

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 88	Tier #		2
Main and Reheat Steam:	Group #		1
Predict the impact of and use procedures to mitigate the	K/A #	039.A2.04	
consequences of a malfunctioning steam dump	Importance Rating		3.7

Proposed Question:
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 failed open.
3. The BOP operator reports that the cause of the failure is Steam Generator Pressure Transmitter PT20D has failed high.
4. The crew enters AOP 3571 *Instrument Failure Response*.
5. The RO reports Calorimetric 4 minute average is 3725 MWth.
6. The US directs the BOP operator to manually CLOSE 3MSS*PV20D.

What other actions is the US required to direct per OP 3204 *At Power Operation* to mitigate the consequences of this event?

A. Immediately reduce power to less than or equal to 100% power, and request Reactor Engineering to determine Reportability.

B. Promptly reduce power to less than or equal to 100% power, and notify the Operations Manager on Call (OMOC).

C. Reduce power to less than or equal to 100% power within 15 to 20 minutes.

D. Monitor power, ensuring it returns to less than or equal to 100% power within 15 to 20 minutes.

Proposed Answer: A

Explanation (Optional): AOP 3571 directs the crew to manually close the failed-open relief valve. OP 3204 actions need to be taken to address the overpower event caused by the malfunctioning steam relief valve. "A" is correct, since these actions are required if 102% power (3723 MWth) is exceeded. "B" is wrong, but plausible, since with power above 100.5% (3668 MWth), the crew is required to promptly reduce power to less than or equal to 100% power, and notify RE. "C" is wrong, but plausible, since this is the action required if power is exceeds 100.2% (3,657 MWth), but not 100.5%. "D" is wrong, but plausible, since, if the 4 minute average power increase was below 100.2% (3657 MWth), the crew would be given 15 to 20 minutes to monitor power fluctuations <100.2% to allow short term transients to dampen out before having to manually reduce power.

Technical Reference(s):	AOP 3571 (Rev. 009-04) Attachment I
(Attach if not previously provided)	OP 3204 (Rev. 017-08), section 1.2, page 3 of 66
(including version/revision number)	OP 3204 (Rev. 017-08), section 4.3.1.a, b, c, and 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: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.1 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 89	Tier #		2
Main Feedwater:	Group #		1
Predict the impact of and use procedures to mitigate the consequences of tripping MFW Pump turbine	K/A #	059.A2.07	
	Importance Rating		3.3

Proposed Question:

Initial Conditions:

- The reactor is at 20% power.
- The "A" TDMFP is running.

The following sequence of events occurs:

1. The crew places the turbine generator on line.
2. A feedwater transient occurs, and all SG NR levels start increasing.
3. "C" SG NR level reaches 85%.
4. The crew desires to reset and start the MDMFP prior to reaching a SG Lo-Lo level trip.

Which procedure is the US required to enter that provides the specific directions on removing the MDMFP trip signal, and what action(s) is/are required to allow resetting and starting the MDMFP?

A. AOP 3550 *Turbine/Generator Trip* provides this direction. The US will allow all SG NR levels to decrease below 80% only.

B. AOP 3550 *Turbine/Generator Trip* provides these directions. The US will allow all SG NR levels to decrease below 80%, and direct the BOP to place the FW Pumps P-4 Trip Bypass Switch to BYPASS on MB5.

C. The SG C LEVEL HI-HI ARP provides this direction. The US will allow all SG NR levels to decrease below 80% only.

D. The SG C LEVEL HI-HI ARP provides these directions. The US will allow all SG NR levels to decrease below 80%, and direct the BOP to place the FW Pumps P-4 Trip Bypass Switch to BYPASS on MB5.

Proposed Answer: A

Explanation (Optional): P-14 trips the main turbine (creating an entry condition into AOP 3550), trips the main feedwater pumps, and causes a Feedwater Isolation. With reactor power less than P-9 (51%), the turbine trip does not result in a reactor trip, so placing the FW Pumps P-4 Trip Bypass Switch to BYPASS on MB5 is not required ("B" and "D" wrong). "B" and "D" are plausible, since above P-9, P-14 would cause P-4 to come in, and the P-4 Trip Bypass Switch would remove this trip from the MDMFP. "A" is correct, since AOP 3550 provides direction on restoring main feedwater flow. "C" is wrong, since the ARP simply directs restore using the Main Feedwater Procedure, which will be too slow to prevent a reactor trip on Lo-Lo SG level. "C" is plausible, since the SG Hi-Hi level annunciator is lit during this event.

Technical Reference(s): AOP 3550 (Rev. 007-04), step 3.a-c.
(Attach if not previously provided) OP 3353.MB5B (Rev. 001-06), 1-5
(including version/revision number) Functional Drawing 13 (Rev. H)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07525 Given a set of plant conditions, determine the required actions to be taken per AOP 3550. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.4, 41.7, and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 90	Tier #		2
AC Electrical Distribution:	Group #		1
Ability to interpret and execute procedure steps	K/A #	062.GEN.2.1.20	
	Importance Rating		4.6

Proposed Question:

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

1. Offsite voltage degrades, followed by a loss of offsite power.
2. The crew enters ECA-0.0 *Loss of All AC Power*.
3. Convex reports the loss of offsite power was momentary, and that offsite power is available.
4. The crew commences ECA-0.0, step 5 "Try to Restore Power to Any AC Emergency Bus."
5. The US directs the BOP to energize bus 34C from offsite power via the RSST.
6. Annunciator BUS 34C UNDERVOLTAGE (MB8A, 3-12) is still LIT.

What action(s) will the applicable procedure direct the operators to take to energize emergency bus 34C from the RSST?

A. The operators will reset the station LOP signal at the sequencer. They will then reset the LOP signal at MB2, and close the RSSA supply breaker to 34C.

B. The operators reset the LOP signal at MB2. They will then reset the station LOP signal at the sequencer, and close the RSSA supply breaker to 34C.

C. The operators will press and hold "BYPASS" on the 34C undervoltage block pushbutton (MB8R). They will then place the RSSA sync selector to ON, and close the RSSA supply breaker to 34C.

D. The operators will press and release "BYPASS" on the 34C undervoltage block pushbutton (MB8R). They will then place the RSSA sync selector to ON, and close the RSSA supply breaker to 34C.

Proposed Answer: C

Explanation (Optional): Choosing the correct procedurally directed action demonstrates proper procedure selection. The correct procedure is GA-3 "Energizing 4.16KV Bus from Offsite Power." GA-3 will direct the operators to press and hold "BYPASS" on the 34C undervoltage block pushbutton (MB8R). Once the LOP occurs, if the RSST does not energize the bus after 1.8 seconds, the RSST supply breaker and the bus tie breaker are locked out for 6 minutes. Prior to the 6 minute timer timing out (as indicated by lit Annunciator BUS 34C UNDERVOLTAGE (MB8A, 3-12), the LOP lockout can only be reset at MB8R if the Pushbutton is held in while off-site power is placed on the bus ("D" wrong). Also, the RSST sync selector switch must be placed to ON to meet the interlock to close the RSST onto the bus ("C" correct). "D" is plausible, since this would be correct if the 6 minute timer has elapsed. "A" and "B" are wrong, since the station LOP reset at the sequencer, and the LOP reset at MB2 removes the Sequencer output signals to plant equipment, restoring manual control from the Main Boards. "A" and "B" are plausible, since the LOP reset pushbuttons on MB2 are used to regain control of equipment during an LOP; and the Station LOP reset pushbutton on the sequencer is operated during LOP restoration steps to reset the sequencer LOP memory, arming it for future LOPs.

Technical Reference(s): ECA-0.0 (Rev. 021), step 5.e and f.
(Attach if not previously provided) GA-3 (Rev. 001), steps 2.b RNO and 2.e.RNO
(including version/revision number)

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… LOP sequence of operations… MB8 alarm response (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.8 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 91	Tier #		2
Pressurizer Level Control: Ability to recognize abnormal parameters that are entry conditions for emergency and abnormal operating procedures	Group #		2
	K/A #	011.GEN.2.4.4	
	Importance Rating		4.7

Proposed Question:

Reactor power is 100% with the pressurizer level control selector switch in the CHAN I-II position when the following **sequence** of events occurs.

1. Letdown automatically isolates and pressurizer heaters trip.
2. Charging flow through 3CHS*FCV121 decreases to zero as indicated on 3CHS*FT121.
3. The crew is slow to respond, and Pressurizer level is at 68% and trending up.

What event is in progress, and which EOP/AOP/ARP will mitigate this event?

A. Pressurizer Level Channel I has failed low. AOP 3571 *Instrument Failure Response* will mitigate this event.

B. Pressurizer Level Channel II has failed low. AOP 3571 *Instrument Failure Response* will mitigate this event.

C. Pressurizer Level Controller 3RCS*LC459 output has failed to 0%. The PRESSURIZER LEVEL DEVIATION ARP will mitigate this event.

D. Charging Line Flow Control Valve 3CHS*FCV121 has failed closed. EOP 3506 *Loss of All Charging Pumps* will mitigate this event.

Proposed Answer: B

Explanation (Optional): Pressurizer low level sensed by either the controlling or backup channel will provide letdown isolation and heater trip protection ("A" plausible). After letdown isolates, charging plus seal injection will cause Pzr level to increase. Since the charging flow controller is sensing the actual increase in level, as evidenced by charging flow decreasing to zero, the backup channel has failed low ("A" wrong), and the correct procedure to mitigate this instrument failure is AOP 3571 ("B" correct). "C" and "D" are wrong, since these failures would not result in letdown isolating until after charging flow has decreased. "C" and "D" are plausible, since charging flow has decreased, and without an instrument failure, the ARP and EOP would apply.

Technical Reference(s): AOP 3571 (Rev. 009-04) Entry Conditions
(Attach if not previously provided) OP 3353.MB4A (Rev. 002-14), 3-1
(including version/revision number) Functional Sheet 11 (Rev. H)

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: Bank #64303

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 # 92		Tier #		2
Fire Protection:		Group #		2
Predict the impact of and use procedures to mitigate the consequences of FPS manual shutdown		K/A #	086.A2.01	
		Importance Rating		3.1

Proposed Question:

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 supplying the electrical auto-close signal to Electro-Thermal Link (ETL) fire damper (3HVR*DMPF142) between the East and West MCC/Rod Control Area is not functional.
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 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.

B. 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.

C. 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.

D. 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.

Proposed Answer: C

Explanation (Optional): 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 ("A" and "B" wrong). "A" and "B" 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. "C" is correct, and "D" 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. "D" 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. 016-07), Section 4.23
(Attach if not previously provided) TRM 3.7.12.3 (March 25, 2004)
(including version/revision number)

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… Evaluate Technical Specification applicability and determine required actions… (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:		Level	RO	SRO
Question # 93		Tier #		2
AMSAC:		Group #		2
Predict the impact of and use procedures to mitigate the consequences of AMSAC malfunction		K/A #	Site-Specific Priority A2	
		Importance Rating		NA

Proposed Question:

With the plant initially at 53% power, the following sequence of events occurs:

1. The plant trips due to a spurious AMSAC actuation signal.
2. A PEO reports that a significant bearing oil leak exists on the running "B" MDAFW pump.
3. Actual SG levels have recovered to 25% NR.
4. The SM contacts the OMOC about the spurious AMSAC actuation.
5. Three minutes after the trip, the spurious AMSAC actuation signal is still present.

What action will the US direct to allow stopping the "B" MDAFW pump?

A. Per ES-0.1 *Reactor Trip Response*, momentarily place the "B" MDAFW Pump control switch on Main Board 5 to STOP, then place the switch in PULL-TO-LOCK (PTL).

B. Per ES-0.1 *Reactor Trip Response*, leave the "B" MDAFW Pump control switch on MB5 in AUTO-AFTER-STOP, wait for 80 seconds to allow AMSAC to dis-arm, and verify the pump trips.

C. Per OP 3350 *ATWS Mitigation System Actuation Circuitry*, place the AMSAC System in BYPASS, and then depress both the Train A and the Train B Steam Generator Low-Low Level Reset pushbuttons on Main Board 5.

D. Per OP 3350 *ATWS Mitigation System Actuation Circuitry*, place the AMSAC System in BYPASS, then momentarily place the "B" MDAFW Pump control switch on MB5 to STOP, and then place the switch in PTL.

Proposed Answer: D

Explanation (Optional): "D" is correct, since bypassing AMSAC per OP 3350 will remove its output signal, allowing manual control. After AMSAC is bypassed, the operator must manually stop the pump. OP 3350 provides specific guidance to bypass AMSAC due to inadvertent actuation. "A" is wrong, since the stop and PTL features are bypassed when AMSAC actuates (ESK 5DY). "A" is plausible, since PULL-TO-LOCK blocks numerous start signals. "B" is wrong, since even though the 260 second timer dis-arms AMSAC, the AFW pump must be taken to STOP before the pump would stop. "B" is plausible, since the 260 second time delay is a dis-arming feature of AMSAC. "A" and "B" are also plausible since a reactor trip has occurred, and the crew has entered ES-0.1. "C" is wrong, since SG Lo-Lo level automatically resets at 2/4 SGs >18% NR, and no further reset is required. "C" is plausible, since the SG Lo-Lo reset pushbuttons also reset the AMSAC start signal, but the switch must be taken to STOP to stop the pump.

Technical Reference(s): OP 3350 (Rev. 006-04), Section 4.3, and Attachment 3.
(Attach if not previously provided) ESK 5DY (Rev. 20)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04090 Given a failure, partial or complete, of AMSAC circuitry, determine the effects on circuit operation and interrelated systems (As available)

Question Source: Bank #77385

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 94	Tier #		3
Knowledge of primary and secondary plant chemistry limits	Group #		1
	K/A #	GEN.2.1.34	
	Importance Rating		3.5

Proposed Question:

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

1. A tube ruptures on the "D" Steam Generator.
2. The crew transitions to E-3 *Steam Generator Tube Rupture*.
3. The CRDSEO (Shift Manager) confers with the ADTS as to which of the following recovery procedures the crew will transition when E-3 is complete:
   - ES-3.1 *Post-SGTR Cooldown Using Backfill*.
   - ES-3.2 *Post-SGTR Cooldown Using Blowdown*.
   - ES-3.3 *Post-SGTR Cooldown Using Steam Dump*.
4. The decision is made to transition to ES-3.1 *Post-SGTR Cooldown Using Backfill*.

In accordance with the WOG Background Document, what is one DISADVANTAGE of using ES-3.1 as the recovery procedure compared to using either ES-3.2 or 3.3?

A. The risk of adverse chemical effects on primary components will be greater.

B. The likelihood of overfilling the ruptured SG will be greater.

C. The potential risk of secondary plant damage due to water hammer will be greater.

D. The radiological release will be greater.

Proposed Answer: A

Explanation (Optional): The major drawbacks of ES-3.1 are related to the fact that secondary plant water will backflow into the RCS. This results in a dilution, and creates the potential for adverse chemical effects on primary components since secondary chemistry limits are not as restrictive as primary chemistry limits. Examples are a tighter pH band for the primary, and the presence of ETA and hydrazine in the secondary coolant ("A" correct). "B" is wrong, but plausible, since overfill is a significant concern that is addressed in E-3, not the ES procedures. "C" is wrong, but plausible, since this is the disadvantage of the ES-3.3 steam dump method. ES-3.1 is generally preferred, since it minimizes rad release ("D" wrong), and facilitates processing of contaminated primary coolant. "D" is plausible, since this is a disadvantage of ES-3.3.

Technical Reference(s): E-3 (Rev. 021), step 41

(Attach if not previously provided) WOG Background Document (Rev. 2) for E-3, step 40

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04373 Discuss conditions which require transition to other procedures from EOP 35 E-3. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 95	Tier #		3
Knowledge of conservative decision making	Group #		1
practices	K/A #	GEN.2.1.39	
Proposed Question:	Importance Rating		4.3

During a storm, screen DP starts increasing on the "A" Circulating Water Pump; and the following sequence of events occurs:

1. The "A" Circulating Water Pump trips.
2. The crew enters AOP 3575 *Rapid Downpower* and starts reducing power at 5%/minute.
3. During the downpower, rods insert below RIL due to Tave/Tref error.
4. The crew commences rapid boration per AOP 3575 *Rapid Downpower*.
5. When power reaches 28%, the STA notices that Tave has lowered below the minimum temperature for criticality.
6. The US directs the RO to pull rods continuously to raise temperature to within the program band.
7. The "B" Circulating Water Pump trips.
8. Condenser backpressure in the "A" condenser bay increases to 6 inches Hg absolute.
9. The crew trips the reactor and enters E-0 *Reactor Trip or Safety Injection*.

What improper action did the US take during this event?

A. The US was required to direct a reactor trip when the "A" Circulating Water Pump tripped.

B. The US was required to enter AOP 3566 *Immediate Boration* when rods inserted below RIL.

C. The US should not have directed the RO to continuously withdraw control rods to restore RCS temperature.

D. The US should have directed a turbine trip and entered AOP 3550 *Turbine Trip*, rather than trip the reactor.

Proposed Answer: C

Explanation (Optional): This event is based on the Salem marsh grass event in SOER 94-1 Non-Conservative Decisions, where operators inappropriately pulled rods continuously with an unstable secondary plant, resulting in a safety injection. "A" is wrong, since the plant is designed to operate with one circ pump running in a bay, and there is no problem with vacuum. "A" is plausible, since Circ Water pumps feed into C-9. "B" is wrong, since AOP 3575 provides adequate guidance for immediate boration with rods below RIL. "B" is plausible, since RIL is normally an entry condition for AOP 3566. "C" is correct, since unexpected reactivity changes shall be thoroughly investigated and resolved, and it is not conservative to add positive reactivity to address unstable plant conditions. "D" is wrong, since, C-9 has been lost, and a reactor trip is required. "D" is plausible, since power is below P-9.

Technical Reference(s): AOP 3575 (Rev. 017-02), Note prior to step 1, and step 7.
(Attach if not previously provided) AOP 3559 (Rev. 009), foldout page.
(including version/revision number) SOER 94-1 Non-conservative Decisions.
OP-AP-300 (Rev. 6), steps 3.7.7, 3.7.12; and Attachment 2, page 4 of 4

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-01925 Demonstrate the ability to make conservative decisions (As available)

Question Source: Bank #77879

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 96	Tier #		3
Knowledge of limiting conditions for operations	Group #		2
and safety limits	K/A #	GEN.2.2.22	
	Importance Rating		4.7

Proposed Question:
Initial Conditions:

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

1. Maintenance reports that they have added the wrong type of oil to the "A" QSS Pump.
2. The crew enters LCO 3.6.2.1 "Containment Quench Spray System," ACTION with one QSS subsystem INOPERABLE, to restore the pump to OPERABLE within 72 hours or be in at least HOT STANDBY within the next 6 hours.
3. The STA looks at the surveillance history of the "B" Train of QSS, and discovers its monthly surveillance valve lineup, scheduled for 20 days ago, was inadvertently missed.
4. The last time the "B" Train QSS lineup was completed was 50 days ago.
5. Work Control estimates the "B" Train valve lineup will be completed in 1.5 hours.

In accordance with section 3/4.0 of Technical Specifications, what ACTION is the crew required/allowed to take?

A. Within one hour, take action to place the unit in HOT STANDBY within the next 6 hours, per LCO 3.0.3, since failure to perform a surveillance within the specified interval shall be failure to meet the LCO, per surveillance requirement 4.0.1.

B. Take action to place the unit in HOT STANDBY within the next 6 hours, per LCO 3.0.3. The one hour allowance of LCO 3.0.3 cannot be utilized, since the estimated time to complete the valve lineup in excess of 1 hour.

C. Remain in the current ACTION for Train "A", since the "B" train surveillance time has not exceeded its maximum allowable extension per surveillance requirement 4.0.2.

D. Remain in the current ACTION for Train "A", since the crew has 24 hours to complete the "B" train surveillance per surveillance requirement 4.0.3.

Proposed Answer: D

Explanation (Optional): "D" is correct, and "A" and "B" wrong, since 4.0.3 allows 24 hours from the time of discovery to complete a missed surveillance prior to declaring the train inoperable. "C" is wrong, since 4.0.2 allows 25% time extension, which has been exceeded 31 days x 1.25 = 38.75 days, and it has been 50 days). "C" is plausible, since 4.0.2 allows an extension for an overdue surveillance. "A" and "B" are plausible, since per 4.0.1, failure to perform a surveillance within the specified interval is failure to meet the LCO, except as provided in 4.0.3.

Technical Reference(s): Tech Spec Section 3/4.0 (Amendments 213 and 241)
(Attach if not previously provided) LCO 3.6.2.1 (Amendment 222)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05790 Describe and apply Technical Specification time interval requirements. (As available)

Question Source: New

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 97	Tier #		3
Knowledge of conditions and limitations in the facility license	Group #		2
	K/A #	GEN.2.2.38	
	Importance Rating		4.5

Proposed Question:

A Steam Generator Tube Rupture occurs at Millstone 3.

What is one "Operator Credited Action" that the FSAR specify for the operators during this event?

A. The crew will verify the PORV Block valve(s) are open within 10 minutes of Safety Injection initiation.

B. The crew will stop AFW Flow to the ruptured Steam Generator by 30% narrow range level.

C. The crew will close the ruptured SG's atmospheric relief valve (assumed to have failed open) within 30 minutes.

D. The crew will terminate Safety Injection within 90 minutes of event initiation.

Proposed Answer: B

Explanation (Optional): "A" is wrong, but plausible; since this operator credited action was recently deleted for an inadvertent SI, where the concern was Pzr overfill. This is no longer an operator credited action due to P-19. "B" is correct, since AFW flow needs to be stopped by 30% Narrow range level (and >8%, to establish the insulating layer). "C" is wrong, but plausible, since the valve needs to be closed within 20 minutes, not 30 minutes as part of the rad release assumptions. "D" is wrong, but plausible, since the time to terminate SIS is 42 minutes, plus assumed times calculated by LOFTTR2 (total time of about 1 hour), which is required to prevent SG overfill. This ensures the Iodine partitioning effect is maintained, keeping rad release within limits.

Technical Reference(s): FSAR (Rev. 21-3), Table 15.6.3-1
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04921 OUTLINE the anticipated Operator Actions in response to SGTRs to include the operator credited actions in FSAR chapter 15. (As available)

Question Source: Bank #71085

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.1 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 98	Tier #		3
Knowledge of normal and emergency exposure limits	Group #		3
	K/A #	GEN.2.3.4	
	Importance Rating		3.7

Proposed Question:

Current Conditions:

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

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

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

A. 4300 mrem TEDE

B. 10000 mrem TEDE

C. 24800 mrem TEDE

D. 25000 mrem TEDE

Proposed Answer: D

Explanation (Optional): Emergency exposure limits for lifesaving or protection of large populations is 25 rem ("D" correct, "A", "B", and "C" wrong). "A" is plausible, since this dose would bring the worker's annual dose to 4.5 rem, which is the maximum TEDE dose allowed for this event without dose extension. "B" is plausible, since this dose is the emergency dose authorized for the protection of valuable property. "C" is plausible, since this dose would bring the worker's dose to 25 rem for the year, which is the emergency limit, but this limit is independent of previous dose.

Technical Reference(s): MP-26-EPI-FAP09 (Rev. 002), Attachment 3.

(Attach if not previously provided)

(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-00688 State the radiation exposure guidelines which have been established for emergencies and the considerations for applying those guidelines. (As available)

Question Source: Bank #74358

Question History: Millstone 3 2004 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 99	Tier #		3
Knowledge of emergency plan protective action recommendations	Group #		4
	K/A #	GEN.2.4.44	
	Importance Rating		4.4

Proposed Question:

The Control Room DSEO has just declared a General Emergency BRAVO.

How will the CRDSEO notify the state of the PAR, and what PAR will be implemented?

A. The PAR will be verbally transmitted to the 24 hour DEP dispatcher in Hartford. State officials will evacuate a 2-mile radius around the site.

B. The PAR will be verbally transmitted to the 24 hour DEP dispatcher in Hartford. An evacuation will NOT be conducted for a General Emergency BRAVO.

C. The Incident Report Form will serve as the PAR notification. State officials will evacuate a 2-mile radius around the site.

D. The Incident Report Form will serve as the PAR notification. An evacuation will NOT be conducted for a General Emergency BRAVO.

Proposed Answer: C

Explanation (Optional): If a General Emergency BRAVO is declared, State officials automatically implement a PAR to evacuate a 2-mile radius ("B" and "D" wrong). The Incident Report Form serves as PAR notification in this instance ("A" wrong and "C" correct). "A" and "B" are plausible, since the PAR will be verbally transmitted to the 24 hour DEP dispatcher in Hartford for General Emergency ALPHA classifications requiring actions out to 10 miles. "B" and "D" are plausible, since on a Site Area Emergency CHARLIE 2, State officials will not conduct an evacuation.

Technical Reference(s): MP-26-EPI-FAP06-005 (Rev. 002), page 1 of 3
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: (As available)

- EP-00203 Explain the method for providing Protective Action Recommendations initially and following activation of the Emergency Response Organization.
- EP-01379 Determine Protective Action Recommendations (PAR) for applicable emergency events.

Question Source: New

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.12 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 100	Tier #		3
Ability to verify alarm setpoints and operate per alarm responses	Group #		4
	K/A #	GEN.2.4.50	
	Importance Rating		4.0

Proposed Question:

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

1. The GEN CORE MONITOR LEVEL HI annunciator comes in on Main Board 7.
2. The dispatched PEO reports back that the local trace shows a drop has occurred from 90% to 5%.
3. The US directs the PEO to depress the "FILTER" pushbutton and report the results.
4. The PEO reports that the trace remains at 5% with the FILTER pushbutton depressed.

What action will the US direct the crew to take?

A. Submit a Condition Report, since indication of core monitor instrument degradation exists.

B. Submit a Condition Report, since indication of core monitor filter clogging exists.

C. Enter AOP 3575 *Rapid Downpower*, and commence a downpower, since indication of Generator overheating exists.

D. Trip the reactor, and go to E-0 *Reactor Trip or Safety Injection*, since indication of Generator overheating exists.

Proposed Answer: A

Explanation (Optional): A decreasing trace occurs if organic material being released into the cooling gas stream, indicating Main Generator overheating is occurring; or if the core monitor instrument is degrading. Placing the filter in service confirms the existence of organic material if the trace recovers with the filter in service, since the filter removes the organic material. However, since the trace did not recover, an instrument failure exists ("A" correct, and "C" and "D" wrong). "C" and "D" are plausible, since a reactor trip would be directed if the trace recovered with the filter in service. "B" is wrong, since a clogged filter would restrict flow, and not allow particulates to pass through. It would be evidenced by low core monitor gas flow with the filter in service. "B" is plausible, since actual generator overheating is not occurring.

Technical Reference(s): OP 3353.MB7C (Rev. 003-05), 4-5
(Attach if not previously provided)
(including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04703 Given a plant condition or equipment malfunction relating to the GMO system, determine when the turbine is required to be tripped, or when the generator must be shutdown. (As available)

Question Source: Modified Bank #74360 Parent Question Attached

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments: