Examination Outline Cross-Reference:	Level	RO	SRO
Question # 1	Tier #	1	1
K/A Statement: Operational implications of decay	Group #	1	1
power as a function of time on a reactor trip	K/A #	EPE.007.EK1.5	
Proposed Question:	Importance Rating	3.3	3.8

The reactor has been at 100% power for several months when the following sequence of events occurs:

1. The reactor trips.
2. The crew enters ES-0.1, *Reactor Trip Response*.
3. One hour after the trip, the BOP operator reports 380 gpm Auxiliary Feedwater flow is required to maintain RCS Tave stable at 557°F.

How long after the trip will it take 150 gpm of AFW flow to maintain RCS Tave stable?

a) 2 to 4 hours.
b) 8 to 12 hours.
c) 1 to 3 days.
d) 7 to 9 days.

Proposed Answer: C

Explanation (Optional): Per GFS thumbrules, decay heat after a trip is 7.5% one second after a trip, 3.75% one minute after a trip, 1.875% one hour after a trip, 0.9375% one day after a trip, and 0.46875% one week after a trip. This is based on starting with 7.5% decay heat after one second, and then cutting the amount of remaining decay heat in half after each increment of time. Since required AFW flow is given as 380 gpm one hour after the trip, cutting 380 gpm in half gives 190 gpm one day after the trip ("A" and "B" wrong), and 95 gpm one week after the trip ("D" wrong). "C" is correct, since 190 gpm is just above the flow required one day after the trip. Per ECA-1.1, Attachment 1, which shows actual required ECCS flow during a LOCA to remove decay heat after a trip, it can be seen that about 380 gpm is required one hour after a trip. About 165 gpm is required after one day (1440 minutes), and about 130 gpm is required after two days. After one week, (10,080 hours) about 90 gpm is required. "A", "B", and "D" are plausible, since these times are all longer than the starting point of one hour after the trip given in the stem, and the required AFW flow of 150 gpm is less than the required 380 gpm one hour after the trip.

Technical Reference(s): ECA-1.1 (Rev 18-0), Attachment A
(Attach if not previously provided, General Physics Reactor Theory Chapter 8 Fig. 8-26 (Rev. 4)
including version/revision number)

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1, 41.5, 41.8, and 41.14

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 2	Tier #	1	1
K/A Statement: Operate/monitor PRT level, pressure	Group #	1	1
and temperature on a Pzr vapor space break	K/A #	APE.008.AA1.8	
Proposed Question:	Importance Rating	3.8	3.8

The plant is in MODE 3, and the following sequence of events occurs:

1. The "A" Pressurizer Safety Valve develops a leak.
2. PRT temperature and pressure start to increase.
3. The PZR REL TK PRESSURE HI annunciator comes in on MB4.
4. The PRT System responds automatically to the high pressure condition.

How will the PRT temperature and pressure trends initially respond when the automatic response occurs?

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): Question is considered a KA match since a leak exists in the Pressurizer through the leaking PORV. "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, or if automatic actions had no effect on these parameters. "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. 05-0), 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 Tanks System controls and interlocks… Pressurizer Relief Tank Vent Valve RCS-PCV469… (As available)

Question Source: Bank #80880

Question History: Millstone 3 2009 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 # 3	Tier #	1	1
K/A Statement: Reasons for actions in the	Group #	1	1
EOP on a small break LOCA	K/A #	EPE.009.EK3.21	
Proposed Question:	Importance Rating	4.2	4.5

Operators are cooling down the plant per ES-1.2, *Post-LOCA Cooldown and Depressurization.*

What RCP configuration does ES-1.2 direct the crew to align, and what is the basis for this configuration?

a) One RCP running for effective heat transfer and pressure control while minimizing heat input.
b) One RCP running for ECCS mixing considerations while minimizing inventory loss.
c) All RCPs stopped to minimize RCS inventory loss and maximize core cooling.
d) All RCPs running, if available, to support a maximum rate cooldown.

Proposed Answer: A

Explanation (Optional): ES-1.2 background states that forced flow is the preferred mode of operation ("C" wrong) to allow for normal RCS cooldown and provide pressurizer spray ("A" correct and "B" wrong). All but one RCP are stopped to minimize heat input to the RCS ("D" wrong). Note that RCP Trip Criteria do not apply once a controlled cooldown is commenced. "B" is plausible, since this is related to the basis for starting a RCP in FR-P.1. "C" is plausible, since this is related to the bases for RCP trip criteria. "D" is plausible, since all RCPs running would provide maximum heat transfer rate from the core to the Steam Generators.

Technical Reference(s): ES-1.2 (Rev. 20-0), step 12
(Attach if not previously provided, ES-1.2 Background Doc (Rev. 3) for RCP major Action
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05350 Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 ES-1.2. (As available)

Question Source: Bank #70250

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 4	Tier #	1	1
K/A Statement: Determine/interpret conditions	Group #	1	1
for throttling/stopping HPI on a large break LOCA	K/A #	EPE.011.EA2.11	
Proposed Question:	Importance Rating	3.9	4.3

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

1. The crew completes aligning for Cold Leg Recirculation per ES-1.3, *Transfer to Cold Leg Recirculation.*
2. The crew returns to E-1, *Loss of Reactor or Secondary Coolant.*

The RO reports the "A" and "B" Containment Recirculation (RSS) Pumps amps and discharge pressure are oscillating.

Based on the recirculation sump criteria of the E-1 foldout page, which of the following choices lists ALL of the pumps required to be tripped AND the correct sequence for tripping those pumps?

a) Trip the "A" and "B" RSS Pumps, then trip both Charging Pumps and both SIH Pumps.
b) Trip the "A" and "B" RSS Pumps, then trip both Charging Pumps, both SIH Pumps, and both RHR Pumps.
c) Trip both Charging Pumps and both SIH Pumps, then trip the "A" and "B" RSS Pumps.
d) Trip both Charging Pumps, both SIH Pumps, and both RHR Pumps, then trip the "A" and "B" RSS Pumps.

Proposed Answer: C

Explanation (Optional): Recirculation Sump Screen Blockage Criteria, which includes RSS Pump amps and flow oscillations, requires prompt stopping of affected pumps receiving suction from the RSS Pump(s), before stopping the associated RSS Pump ("A" and "B" wrong). "C" is correct, and "D" wrong, since RHR Pumps are not required to be stopped, since they are removed from service when aligning for Cold Leg Recirculation. "A" and "B" are plausible, since all of these pumps are required to be tripped, just the sequence is wrong. "D" is plausible, since the RHR pumps initially supply low head injection, and are stopped as part of aligning for recirculation.

Technical Reference(s):	E-1 (Rev 26-0), Foldout Page
(Attach if not previously provided,	P&ID 112C (Rev. 38)
including version/revision number)	P&ID 112A (Rev. 50)
	P&ID 104A (Rev. 54)
	P&ID 113B (Rev. 42)

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 5	Tier #	1	1
K/A Statement: Determine/interpret the cause	Group #	1	1
of RCP failure	K/A #	APE.015.AA2.1	
Proposed Question:	Importance Rating	3.0	3.5

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

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

Lubrication will be lost due to___(1)___leaking into the___(2)___.

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

Proposed Answer: C

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

Technical Reference(s): OP3353.MB4B (Rev. 08-0), 4-2A
(Attach if not previously provided, P&ID 102A (Rev. 33)
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 #89256

Question History: Millstone 3 2013 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3 and 41.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 6	Tier #	1	1
K/A Statement: Operational implications of the reason for changing from manual to automatic control of charging flow during a loss of makeup	Group #	1	1
	K/A #	APE.022.AK1.4	
	Importance Rating	2.9	3.0

Proposed Question:

With the plant initially at 50% power, the following sequence of events occurs:

1. The "A" Charging Pump trips.
2. The RO isolates Charging and Letdown per AOP 3581, *Immediate Operator Actions.*
3. The crew enters AOP 3580, *Loss of Charging Pumps.*
4. The RO attempts to start the "B" Charging Pump, and it fails to start.
5. The crew successfully aligns and starts the "C" Charging Pump and restores letdown.
6. The RO begins restoring Pressurizer level to normal using MANUAL control of Charging Flow Control Valve 3CHS*FCV121.

Assuming no further adjustments are made with 3CHS*FCV121, and the controller is left in manual, select the first significant automatic action that will occur from among the following choices.

a) Letdown will isolate on low Pressurizer level.
b) The Reactor will trip on low Pressurizer pressure.
c) The Reactor will trip on high Pressurizer pressure.
d) The Reactor will trip on high Pressurizer level.

Proposed Answer: D

Explanation (Optional): With all Charging Pumps tripped, Pzr level will begin to decrease, even with Letdown isolated, due to RCP seal leakoff flow without charging flow or seal injection. While the "C" Charging Pump is being aligned (which takes time), Pzr level will drop below program level. The RO initially fully closed the Charging Flow Control Valve, and since level is low, the RO throttles FCV 121 open to restore level. This means there is a net gain in RCS inventory, causing Pzr level to increase ("A" wrong). Without operator action, eventually a high pressurizer level reactor trip will occur ("D" correct). "B" is wrong, since raising Pzr level squeezes the bubble in the Pzr, raising RCS pressure. "B" is plausible, since if the controller was left in manual with net charging plus seal injection less than net letdown plus seal leakoff flow, Pzr level would decrease, expanding the bubble, lowering Pzr pressure and level (making "A" plausible). "C" is wrong, but plausible based on pressurizer pressure rising in response to rising pressurizer level. However, the 2385 psia Reactor trip would not be met due to the expectant response of the Pressurizer Spray Valves.

Technical Reference(s): AOP 3580 (Rev 03-0), steps 7 d, g, h, and step 11
(Attach if not previously provided, Functional sheet 11 (Rev. H)
including version/revision number)

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 7	Tier #	1	1
K/A Statement: Operate/monitor LPI pumps	Group #	1	1
on a loss of Residual Heat Removal	K/A #	APE.025.AA1.3	
Proposed Question:	Importance Rating	3.4	3.3

Initial conditions:

- The plant is in MODE 5.
- Both trains of RHR are operating in the "Cooldown" Mode.

The following sequence of events occurs:

1. Pzr level starts decreasing.
2. The crew enters EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory.*
3. Per EOP 3505, the US directs the RO to align RHR Train B for shutdown risk inventory control to restore Pzr level.
4. The RO places the "B" RHR Pump in Pull-To-Lock.

Which, if any, of the following two valves will the RO be required to reposition prior to restarting the "B" RHR Pump?

- 3SIL*MV8809B, "PP B COLD LEG INJ"
- 3SIL*MV8812B, "RWST/PP B SUCT ISOL"

a) Neither valve is required to be repositioned.
b) Only 3SIL*MV8809B needs to be repositioned.
c) Only 3SIL*MV8812B needs to be repositioned.
d) Both valves are required to be repositioned.

Proposed Answer: C

Explanation (Optional): This question is considered a KA match since at Millstone 3, the RHR Pumps also function as the Low Pressure Safety Injection Pumps. The crew will be realigning Train B RHR to inject water from the RWST into the RCS Cold Legs. The B RHR Cold Leg Injection Valve is already open in the Cooldown Mode, so opening the valve is not required. With an actual loss of inventory in progress, the crew will be directed to ensure the Cold Leg Injection Valve is open, since it is already open in the cooldown mode ("B" and "D" wrong). The suction path is initially aligned to take a suction on the RCS in the Cooldown Mode, so the RWST Suction Valve is required to be opened ("C" correct, "A" wrong). "A", "B", and "D" are plausible, since these valves may be open or closed, depending on the event in progress (Cooldown, Injection, Cold Leg Recirc).

Technical Reference(s): (Attach if not previously provided, including version/revision number)
EOP 3505 (Rev 15-0), Entry Condition B.2.a, and step 5.RNO.e.2)
OP 3310A (Rev 18-0), Steps 4.22.7.a and 4.22.8
P&ID 112A (Rev. 50)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05451 Describe the function and location of the following Residual Heat Removal System components... RWST Suction Valves... RHR Pumps ... RHR to RCS Cold Leg Injection Valves... (As available)

Question Source: New

Question History:

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 # 8	Tier #	1	1
K/A Statement: Knowledge of system purpose	Group #	1	1
or function during a loss of component cooling water	K/A #	APE.026.GEN.2.1.27	
Proposed Question:	Importance Rating	3.9	4.0

With the plant at 100% power and the "C" RPCCW Pump unavailable, the following sequence of events occurs:

1. The "B" RPCCW Pump trips and cannot be restarted.
2. The crew enters AOP 3561, *Loss of Reactor Plant Component Cooling Water*.
3. The crew performs the appropriate actions of AOP 3561 to mitigate the event.

Complete the following statement concerning which components no longer have RPCCW available.

The __(1)__ Heat Exchanger is NOT receiving cooling water from RPCCW, and the __(2)__ Cooling Surge Tank does NOT have makeup water available from RPCCW.

	(1)	(2)
a)	Seal Return	Charging Pumps
b)	Excess Letdown	Charging Pumps
c)	Seal Return	Safety Injection Pumps
d)	Excess Letdown	Safety Injection Pumps

Proposed Answer: A

Explanation (Optional): The Seal Return Heat Exchanger and the Charging Pumps Cooling Surge Tank are serviced by the "B" train of RPCCW Containment Header, but are outside of CTMT. Thus when the CTMT headers are cross-connected (which includes closing the "B" train CTMT Isolation Valves), these components will be affected ("A" correct). The Excess Letdown Heat Exchanger is a "B" train RPCCW load, but it is located inside CTMT and will have flow once CTMT headers are cross-connected ("B" and "D" wrong but plausible). The Safety Injection Pump Cooling Surge Tank is supplied by "A" Train RPCCW ("C" and "D" wrong, but plausible).

Technical Reference(s): AOP 3561 (Rev. 18-0), Attachment A, page 1
(Attach if not previously provided, P & ID 121A (Rev. 33)
including version/revision number) P & ID 121B (Rev. 21)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04155 (215500); Given the following failures, partial or complete, determine the effects on the RPCCW System, and on interrelated systems .... RPCCW leak .... High motor temperature (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4 and 41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 9	Tier #	1	1
K/A Statement: Interrelations between controllers/positioners	Group #	1	1
and a pressurizer pressure control malfunction	K/A #	APE.027.AK2.3	
Proposed Question:	Importance Rating	2.6	2.8

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

The Master Pressurizer Pressure Controller malfunctions and the **setpoint** step-changes from 2250 psia to 2175 psia, and components reposition.

After placing the Pressurizer Master Pressure Controller to MANUAL, what action will the RO take with the Master Pressurizer Pressure Controller in response to the failure?

a) Push on the INCREASE pushbutton, closing both spray valves and energizing backup heaters "A","B", "D", and "E".
b) Push on the DECREASE pushbutton, closing both spray valves and energizing backup heaters "A","B", "D", and "E".
c) Push on the INCREASE pushbutton, deenergizing backup heaters "A","B", "D", and "E", and opening both spray valves.
d) Push on the DECREASE pushbutton, deenergizing backup heaters "A","B", "D", and "E", and opening both spray valves.

Proposed Answer: A

Explanation (Optional): With the controller attempting to maintain 2175 psia, spray valves will have opened and heaters will have deenergized. Actual pressure will have started dropping, so the RO will need to close spray valves and energize heaters ("C" and "D" wrong). "C" and "D" are plausible since this would be the response if the controller had failed in the other direction. To raise pressure, the RO needs to go in the INCREASE direction ("A" correct, "B" and "D" wrong). "B" and "D" are plausible since this would be true of selecting INCREASE would OPEN on the spray valves.

Technical Reference(s): Functional Sheet 11 (Rev. H)
(Attach if not previously provided, Functional Sheet 12 (Rev. F)
including version/revision number) Process Sheet 26 (Rev. J)

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 #78908

Question History: Millstone 3 2004 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 10	Tier #	1	1
K/A Statement: Interrelations between	Group #	1	1
breakers/relays/disconnects and an ATWS	K/A #	EPE.029.EK2.6	
Proposed Question:	Importance Rating	2.9	3.1

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

1. A steam pipe breaks in Containment.
2. SI and MSI automatically actuate.
3. The reactor fails to trip and the crew enters EOP 35 FR-S.1, *Response to Nuclear Power Generation / ATWS*.
4. After progressing through the EOP network, the crew transitions to ES-1.1, *SI Termination*.
5. When the Reactor Operator resets SI, it immediately re-actuates.

What error by the crew could be the cause of the SI reset failure?

a) The crew has NOT waited for the SIS initiating condition to clear.
b) The crew has NOT opened the Reactor Trip Breakers.
c) The crew has NOT allowed the SI Block/Reset logic to time out.
d) The crew has NOT lowered RCS pressure below the P-11 setpoint.

Proposed Answer: B

Explanation (Optional): When SIS actuates, there is a 60 second time delay that must time out before SIS can be reset. "B" is correct since one function of P-4 is to prevent automatic SI actuation after SI is reset. "A" is wrong, since SI can be reset with the actuating signal still present if P-4 is also present, and greater than 60 seconds has elapsed. It is not desired to restore the faulted SG pressure >660 psig prior to resetting SI. "A" is plausible, since the actuating signal is still present. "C" is wrong, since the procedure flowpath to get to ES-1.1 would be significantly longer than 60 seconds. "C" is plausible, since a time delay exists before reset is possible. "D" is wrong, since P-11 does not have to be energized to allow SI reset. "D" is plausible, since P-11 provides the ability to block the Low Pressurizer Pressure SI at pressures below 1950 psig.

Technical Reference(s): Functional Sheet 8 (Rev. K)
(Attach if not previously provided,
including version/revision number)

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... Instrumentation Failure... (As available)

Question Source: Bank #72376

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 # 11	Tier #	1	1
K/A Statement: Operational implications of	Group #	1	1
leak rate versus pressure drop on a SGTR	K/A #	EPE.038.EK1.2	
Proposed Question:	Importance Rating	3.2	3.5

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 takes actions to identify and isolate the affected SG and cooldown the RCS.
3. The crew reaches step 16 which directs RCS depressurization.
4. The crew commences depressurizing the RCS using maximum normal Pzr spray.
5. The crew closes the normal PZR spray valves when the RCS pressure termination criterion is reached.
6. Immediately after the PZR Spray Valves are closed, the US directs the RO to monitor Pzr pressure and level.

Assuming no other pipe breaks or equipment malfunctions exist, complete the following statement.

The RO reports Pzr pressure and level are ______________.

a) increasing, since Safety Injection flow has not yet been terminated
b) increasing, since the RCS is heating up
c) decreasing, since primary to secondary leakage will reinitiate
d) decreasing, 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 lowers RCS pressure, decreasing break flow and increasing injection flow. The goal of the depressurization is to drop RCS pressure less than ruptured SG pressure. Unlike a LOCA, which will continue until the RCS is completely depressurized, RCS leak rate will decrease and then stop when RCS is less than ruptured SG pressure. Reverse flow from the ruptured SG into the RCS will commence. This is a temporary stop in primary to secondary leakage, since SI is still injecting, and with mass of the RCS increasing, the RCS will repressurize and break flow out the RCS into the SG will recommence. "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. 25-0), steps 6.f, 13.c, 16, 18, and 19.
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: ______________

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

Question Source: Bank #85230

Question History:

Question Cognitive Level: Comprehension or Analysis

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 12	Tier #	1	1
K/A Statement: Reasons for actions in EOPs	Group #	1	1
for a loss of Main Feedwater	K/A #	APE.054.AK3.4	
Proposed Question:	Importance Rating	4.4	4.6

With the plant operating at 100% power, the Reactor trips due to a loss of both Turbine Driven Main Feedwater Pumps (TDMFPs), resulting in the following sequence of events:

1. Safety Injection actuates.
2. The crew transitions from E-0 to FR-H.1, *Response to Loss of Secondary Heat Sink.*
3. The crew commences efforts to establish Main Feed flow from the Motor Driven Main Feedwater Pump (MDMFP).
4. The crew resets Safety Injection.
5. The crew removes the Universal Logic (A213) Cards from 3RPS*RACLOGA and B.
6. The crew resets FWI at MB2 and MB5, and opens the FW Isolation Trip Valves.

Complete the following statement concerning why FR-H.1 directed the crew to remove the universal logic cards as part of the efforts to restore Main Feedwater flow.

Removing the Universal Logic Cards removes a seal-in feature generated from a__(1)__, allowing the crew to__(2)__.

	(1)	(2)
a)	Safety Injection plus P-4 signal	start the MDMFP
b)	Safety Injection plus P-4 signal	open the Feed Reg Bypass Valves
c)	P-4 plus Low Tave signal	start the MDMFP
d)	P-4 plus Low Tave signal	open the Feed Reg Bypass Valves

Proposed Answer: B

Explanation (Optional): SIS, P-14, or P-4 with Lo Tave ("C" and "D" plausible) generate a FWI signal. The SIS signal combines with the P-4 signal to lock in the FWI. The universal logic cards must be pulled to clear the seal in feature of the FWI circuit, which prevents opening the Feed Reg Valves, Feed Reg Bypass Valves ("B" correct), and Feed Isolation Trip Valves. This seal in feature does not affect the ability to reset the Main Feed Pumps as the SIS signal feeds into the Pump trip upstream of this part of the circuit (see Functional Sheet 13) ("A" wrong, but plausible). "C" and "D" are wrong, since P-4 plus Lo Tave portion of the FWI circuit is electrically in parallel with the universal logic card, allowing this portion of the FWI signal to be reset even with the P-4 plus Lo Tave condition still present.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	FR-H.1 (Rev. 26-0), step 6.b.RNO Functional Sheet 13 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04535 Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 FR-H.1…	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 13	Tier #	1	1
K/A Statement: Determine/interpret AFW	Group #	1	1
flow indicator on a loss of offsite power	K/A #	APE.056.AA2.20	
Proposed Question:	Importance Rating	3.9	4.1

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

1. The reactor trips due to a loss of offsite power.
2. The crew enters ES-0.1, *Reactor Trip Response.*
3. The US directs the BOP to maintain total feed flow between 530 and 600 gpm until narrow range level is greater than 8% in at least one SG.

Complete the following statements.

Prior to the BOP taking action, the expected AFW flowrate to each SG is __(1)__ gpm. With the TDAFW Pump feeding the SGs, the BOP operator is restricted in full travel stroking of AFW Flow Control Valves to one at a time, at a rate greater than __(2)__ seconds over full travel.

	(1)	(2)
a)	300	15
b)	400	15
c)	300	5
d)	400	5

Proposed Answer: A

Explanation (Optional): The Cavitating Venturis limit AFW flow to about 300 gpm per SG. OP 3272, *EOP/AOP User's Guide,* states that "With the TDAFW Pump feeding forward and when throttling TDAFW or MDAFW Flow Control Valves either OPEN or CLOSED, the valves should be throttled one at a time, at a rate that is greater than 15 seconds over the full travel." This is due to the low margin between the operating pressure of the TDAFW pump and the discharge relief valve setpoint. Therefore, "A" is correct. "B" and "D" are plausible, since 400 gpm per SG is the approximate design flow of the TDAFW pump plus one MDAFW pump.

Technical Reference(s): ES-0.1 (Rev. 029-0), step 1
(Attach if not previously provided, OP 3272 (Rev. 09-0) Attachment 3, page 39
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04454 (215730); State the conditions which would allow… throttling of or isolation of auxiliary feed water flow to a steam generator… prior to being directed to perform the action by a specific step within the EOP network. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5 / 45.13

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 14	Tier #	1	1
K/A Statement: Ability to determine operability/availability of safety related equipment on a loss of vital AC instrument bus	Group #	1	1
	K/A #	APE.057.GEN.2.2.37	
	Importance Rating	3.6	4.6

Proposed Question:

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

1. VIAC 2 deenergizes.
2. The crew enters AOP 3564, *Loss of VIAC*.
3. A loss of offsite power occurs while VIAC 2 is de-energized.

Complete the following statement concerning automatic starting of the "B" EDG, and Bus 34D load stripping.

The "B" EDG __(1)__, and loads on Bus 34D __(2)__ strip.

	(1)	(2)
a)	does NOT start	DO NOT
b)	does NOT start	DO
c)	starts, but its output breaker does NOT close	DO NOT
d)	starts, but its output breaker does NOT close	DO

Proposed Answer: C

Explanation (Optional): A loss of VIAC- 2 deenergizes the "B" EDG Sequencer. The loss of a sequencer with an ESF Actuation Signal present results in the following:
The associated diesel will not start ("A" and "B" plausible) except on a LOP, due to a start signal directly from Undervoltage relays on the bus) ("A" and "B" wrong).
The sequencer normally sends a "Bus stripped" auto-close permissive signal to the EDG output breaker. Without the signal, the breaker will not close.
Loads will not be stripped from the emergency bus ("C" correct, "D" wrong). "D" is plausible, since normally, loads strip on an LOP.
Load sequencing will not occur if the bus is energized.

Technical Reference(s): (Attach if not previously provided, including version/revision number) AOP 3564 (Rev. 11-0), Caution prior to step 1
LSK-24-9.3A (Rev. 9)
LSK-24-9.4A (Rev. 12)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07548 Given a set of plant conditions, properly apply the notes and cautions of AOP 3564. (As available)

Question Source: Bank #69207

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 41.7, 41.7, 41.8, and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 15	Tier #	1	1
K/A Statement: Reasons for conditions that initiate automatic	Group #	1	1
opening/closing of SWS isolation valves to coolers	K/A #	APE.062.AK3.1	
Proposed Question:	Importance Rating	3.2	3.5

Why are Service Water Supply to RPCCW Valves 3SWP*MOV50A and B designed to automatically close on a CDA Signal?

a) Allow adequate pressure to refill the Control Building Chiller Service Water Booster Pump suction piping.
b) Allow adequate pressure to refill the MCC/Rod Control Area Service Water Booster Pump suction piping.
c) Prevent robbing flow from the EDG Service Water Coolers in the event of an LOP.
d) Prevent excessive flow conditions in the Service Water System while supplying RSS.

Proposed Answer: D

Explanation (Optional): "D" is correct, and "A", "B", and "C" 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 total flow to RPCCW Heat Exchangers is 14,776 gpm (7388 gpm per train), and flow to a train of RSS Heat Exchangers is 21,600 gpm (10,800 gpm per train. So with both the RPCCW Heat Exchangers and the RSS Heat Exchangers being supplied by a train on SWP, flow from that train would be 18,188 gpm, which exceeds the maximum flow for a SWP Pump. "A" and "B" are plausible, since both the Control Building Chiller Booster Pumps and the MCC/Rod Control Booster Pumps are at a high elevation, and 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). "C" is plausible, since the EDG cooling water valves automatically open on a CDA, to provide cooling to the EDGs, which automatically started.

Technical Reference(s):	OP 3326 (Rev. 31-0), Precaution 3.7
(Attach if not previously provided,	FSAR Table 9.2-1 (Rev. 30)
including version/revision number)	P &ID 133A (Rev. 44-0)
	P &ID 133B (Rev. 89-0)

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 (SWP*MOV50A/B)... (As available)

Question Source: Bank #85235

Question History: Millstone 3 2009 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 16	Tier #	1	1
K/A Statement: Operate/monitor the emergency	Group #	1	1
air compressor on a loss if instrument air	K/A #	APE.065.AA1.4	
Proposed Question:	Importance Rating	3.5	3.4

The plant is operating normally at full power when the following sequence of events occurs.

1. The RO observes instrument air header pressure at 90 psig and slowly decreasing.
2. The crew enters AOP 3562, *Loss of Instrument Air.*
3. The RO observes that the Diesel Instrument Air Compressor did not auto start and a PEO is dispatched to place this compressor in service.
4. The PEO places the Diesel Instrument Air Compressor control switch on 3IAS-PNL3 to MANUAL.
5. One minute later the PEO observes Diesel Instrument Air Receiver pressure indicator 3IAS-PI97 unchanged, indicating 0 psig and steady.

What is the expected PEO action to comply with AOP 3562?

a) Press the Diesel Air Compressor AUTO MODE pushbutton on 3IAS-PNL3.
b) Place the Diesel Instrument Air Compressor switch on 3IAS-PNL3 to OFF.
c) Continue monitoring for rising air pressure on pressure indicator 3IAS-PI97.
d) Check the Diesel Air Compressor fuel tank level indicating greater than 91%.

Proposed Answer: C

Explanation (Optional):
The PEO is dispatched with air pressure at 90 psig, which is well above the auto-start setpoint of 75 psig. The Diesel Instrument Air System manual start sequence from 3IAS-PNL3, has time delays which control and start the associated compressor and dryer. When the control switch is placed to "MANUAL," a three minute time delay will start and ensures that the Diesel Generator has time to start, power the compressor and dryer PLCs, and then initiate a start signal to the compressor ("C" correct, "A", "B", and "D" wrong). "A", "B", and "D" are plausible, since pressure is not increasing. "A" is wrong since this is the normal control mode configuration and the compressor will not start until 75 psig is reached (& it's desired to start the compressor now). "B" is wrong as placing the compressor control switch in OFF will disable the compressor. "D" is wrong as there is no fuel tank level interlock preventing compressor start.

Technical Reference(s): AOP 3562 (Rev. 16-0), step 2. RNO.b.1
AOP 3562 (Rev. 16-0), Attachment B
AOP 3562 background doc (Rev. 16-0) for Attachment B, Note prior to step B.1

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05319 (216547): Describe the function and location of the following Plant Air System components… Diesel Instrument Air Compressor Subsystem. (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 # 17	Tier #	1	1
K/A Statement: Interrelations between heat removal systems,	Group #	1	1
proper operations of those systems, and a loss of secondary	K/A #	WE05.EK2.2	
heat sink	Importance Rating	3.9	4.2

Proposed Question:

The crew has entered FR-H.1, *Response to Loss of Secondary Heat Sink*, and current conditions are as follows:

- The crew has NOT yet been required to initiate Bleed and Feed cooling of the RCS.
- The crew is preparing to depressurize Steam Generators to attempt to establish feed flow from the Condensate System.

In accordance with FR-H.1, what is the minimum number of Steam Generators the crew is required to depressurize, and to what minimum pressure are they required to be depressurized to?

	Minimum number of SGs	Minimum SG pressure
a)	1 or 2	400 psig
b)	1 or 2	800 psig
c)	3 or 4	400 psig
d)	3 or 4	800 psig

Proposed Answer: A

Explanation (Optional): To establish Condensate Pump flow to the Steam Generators, the crew is directed to depressurize SGs to a pressure that will allow condensate flow to the Steam Generators. The crew is allowed to depressurize one or two SGs, rather than all four, since this reduces the likelihood of reaching Bleed and Feed criteria (three of four SGs less than 21% WR level). This is allowed since steaming SGs to lower their pressure also lowers their WR level ("C" and "D" wrong). "C" and "D" are plausible, since depressurizing all four SGs would cool the RCS, and provide a greater chance of success if bleed and feed were not a concern. The minimum SG target pressure is 400 psig ("A" correct, "B" wrong), since this pressure ensures a Condensate Pump can provide adequate flow for RCS removal, and also have adequate recirculation flow for pump cooling. "B" is plausible, since this is well below normal SG pressure of 1092 psig after a trip.

Technical Reference(s): FR-H.1 (Rev 26-0), step 9.b and f RNO
(Attach if not previously provided, FR-H.1 Millstone EOP Bkgd Document (Rev 26-0), step 9
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04534 Describe major action categories within EOP 35 FR-H.1 (As available)

Question Source: New

Question History:

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 # 18	Tier #	1	1
K/A Statement: Local Auxiliary Operator tasks during	Group #	1	1
a steam line rupture, and resultant operational effects	K/A #	WE12.GEN.2.4.3	
Proposed Question:	Importance Rating	3.8	4.0

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

1. Safety Injection actuates.
2. The BOP Operator reports that the Turbine Driven AFW Pump is running.
3. The BOP Operator reports the Motor Driven AFW Pumps will NOT start.
4. The crew identifies all SGs are faulted and enters ECA-2.1, *Uncontrolled Depressurization of All Steam Generators*.
5. A PEO is provided a copy of ECA-2.1, Attachment B, "Guidance for the Local Isolation of a Faulted SG", and is directed to attempt to close ALL valves listed in the attachment.

Assuming the PEO closes all of the valves on Attachment B, what effect(s), if any, will the PEO's actions have on the TDAFW Pump and associated AFW flowpaths, and why?

a) There will be NO effect, since ECA-2.1, Attachment B focuses on isolating the major steam paths and feed paths from the four faulted SGs.
b) The pump will remain running BUT AFW flow to the SGs will stop because the pump discharge paths will be isolated.
c) The AFW flowpaths will remain aligned BUT the pump will stop because the turbine steam supply paths will be isolated.
d) The pump will stop AND the AFW flowpaths will not be aligned because the turbine steam supplies AND the pump discharge paths will be isolated.

Proposed Answer: D

Explanation (Optional): Attachment B isolates the major paths from the SGs, such as the main steamline and main feedline ("A" plausible), and also isolates smaller paths, down to the Chemical feed lines. It isolates the AFW Pump discharge paths ("A" and "C" wrong, "B" plausible), and the TDAFW Pump turbine steam supply valves ("D" correct, "B" wrong, "C" plausible). The crew is required to brief the PEO to ensure only the desired valves will be closed based on the event in progress.

Technical Reference(s): ECA-2.1 (Rev 19-0), Attachment B
(Attach if not previously provided, P&ID 130B (Rev. 49)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04367 Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 E-2. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.8 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 19	Tier #	1	1
K/A Statement: Operate/monitor facility behavior	Group #	2	2
characteristics during an instrument failure	K/A # (Site Priority)	AOP 3571.AA1.2	
Proposed Question:	Importance Rating	3.3	3.6

The plant is in MODE 4, COPPS is armed and both PORVs are in AUTO, when the plant experiences a failure of the RCS Loop 1 Wide Range Thot instrument.

Which of the following correctly describes the effect on the PORV pressure lift setpoint as compared to immediately prior to the instrument failure?

a) The train "A" PORV will open at a higher pressure setpoint if the Thot instrument has failed high.
b) The train "B" PORV will open at a higher pressure setpoint if the Thot instrument has failed high.
c) The train "A" PORV will open at a lower pressure setpoint if the Thot instrument has failed low.
d) The train "B" PORV will open at a lower pressure setpoint if the Thot instrument has failed low.

Proposed Answer: C

Explanation (Optional): WR Thot inputs to PORV "A", and WR Tcold inputs to PORV"B" ("B" and "D" wrong, but plausible). The input is auctioneered LOW. If the temperature input fails low, the calculated pressure setpoint will be lower ("C" correct), since at colder temperatures, the vessel is more susceptible to an overpressure event. "A" is wrong, but plausible, since the circuit looks at auctioneered low Thot.

Technical Reference(s): OP 3208 (Rev. 27-0), Attachment 1
(Attach if not previously provided, Tech Spec Figures 3.4-4.A and B (Amendment 197)
including version/revision number) Functional Sheet 18 (Rev. D)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05447 Given a failure, partial or complete, of the Reactor Coolant System, DETERMINE the effects on the system and on interrelated systems. (As available)

Question Source: Bank #68333

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 # 20	Tier #	1	1
K/A Statement: Reasons for guidance contained	Group #	2	2
in the EOP for a fuel handling incident	K/A #	APE.036.AK3.3	
	Importance Rating	3.7	4.1

A fuel handling accident occurs in the Fuel Building, resulting in the following sequence of events:

1. The crew enters EOP 3502, *Fuel Handling Accident.*
2. Per EOP 3502, step 11, the crew is verifying that a Fuel Building Filter Unit (3HVR*FN10A or B) is running.

Why 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 (Rev. 09-0), step 11
(Attach if not previously provided,	EOP 3502 Basis Doc (Rev. 09-0), step 11
including version/revision number)	FSAR Section 9.4.3 (Rev. 30)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-06415 Discuss the bases of major procedure steps and/or sequence of steps in EOP 3502, Fuel Handling Accident	(As available)

Question Source:	Bank #85241
Question History:	Millstone 3 2009 NRC Exam
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 # 21	Tier #	1	1
K/A Statement: Evaluate plant performance and make	Group #	2	2
operational judgments during an accidental liquid	K/A #	APE.059.GEN.2.1.7	
radwaste release	Importance Rating	4.4	4.7

Proposed Question:

A RADIATION HI annunciator is received on MB2.

A PEO reports the Turbine Plant Component Cooling Water (TPCCW) Sump level has started increasing.

Radiation detected from which location could be causing this abnormal level increase in the TPCCW sump?

a) Turbine Building Floor Drain Sump
b) SG Blowdown Tank
c) Waste Neutralization Sump
d) Liquid Waste (LWS) Discharge Piping

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since only the Turbine Building Floor Drains Sump Radiation Monitor RE50 diverts flow to the TPCCW sump upon detection of Hi Radiation levels. "B" is plausible, since the Blowdown Tank is a potential liquid release source in the event of a SG Tube Leak, and the blowdown line has an associated radiation monitor. "C" is plausible, since Radiation Monitor 3CND-07 diverts to the Condensate Demin Waste Neutralizing Sump on high radiation. "D" is plausible, since the TPCCW sump is normally aligned to the Liquid Waste System.

Technical Reference(s): AOP 3573 (Rev. 23-0), Att. A, pages 1, 3, 4, and 11 of 12
(Attach if not previously provided,
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.. DAS-RE50... (As available)

Question Source: Bank #67219

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5, 41.7, and 41.11

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 22	Tier #	1	1
K/A Statement: Operate/monitor facility	Group #	2	2
behavior characteristics during a turbine trip	K/A # (Site Priority)	AOP 3550.AA1.2	
Proposed Question:	Importance Rating	3,3	3.6

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 RO reports Tave has peaked at 574°F and is lowering.
4. At 30% power, the RO places Rod Control in Manual.

Assuming the reactor does NOT trip, complete the following statements.

After the turbine tripped, __(1)__ of the Condenser Steam Dump Valves fully opened. After rods were placed in manual, Condenser Steam Dump Valves demand stabilized at about __(2)__ demand.

	(1)	(2)
a)	some but NOT all	60%
b)	some but NOT all	75%
c)	ALL	60%
d)	ALL	75%

Proposed Answer: D

Explanation (Optional): On a turbine trip, a C-7 signal (Load Reject) is generated after a 10% step decrease in turbine load (this has occurred). This arms and opens condenser steam dump valves. Maximum steam dump capacity is 40% reactor power, so they initially all fully open ("A" and "B" wrong). "A" and "B" are plausible, since on a load reject, rods start automatically inserting, and they have the capacity to compensate for a 10% load reject. This is caused by a primary to secondary power rate of change signal, along with Tave above Tref. Tave starts increasing on the turbine trip (heat in from the reactor is greater than heat out). Rods start to automatically driving in. Tref dropped to no-load Tave (557°F) on the turbine trip. On the turbine trip, with Tref dropping to 557°F, and Tave initially increasing, the Tave-Tref error is instantly 12°F high (Tref at initial power of 40% is 569°F), and starts rapidly increasing. The temperature error exceeds the 15.8°F trip open setpoint (Validated on the simulator), based on Tave peaking at 574°F. The steam dumps have a 2°F deadband, and rods have a 1.5°F deadband. If both systems are left in auto, rods will continue to step in to lower temperature to within 1.5°F of program, closing the steam dumps and reducing power to below the point where Main Feedwater Control Valves can control SG levels. Rods are placed in manual to stabilize reactor power, after which the crew can lower power in a controlled fashion. In this case, rods were placed in manual at 30% reactor power, which is about 75% demand, since its full capacity is 40% power ("D" correct, "C" wrong). "C" is plausible, since reactor power is 30%, and SG Safety Valves are able to respond to a 100% load reject.

Technical Reference(s):	AOP 3550 (Rev 10-0), step 2.c and 2.d
(Attach if not previously provided,	AOP 3550 Basis Doc (Rev 10-0), step 2
including version/revision number)	Functional Sheets 9 (Rev. H) and 10 (Rev. J)
	SDS041C (Rev. 5), Section 3

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1, 41.5, 41.6, 41.7, and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 23	Tier #	1	1
K/A Statement: Interrelations between components and	Group #	2	2
functions of control and safety systems, including instru-	K/A #	WE01 & 02.EK2.1	
mentation, interlocks, failure modes during SI termination	Importance Rating	3.4	3.9

Proposed Question:

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

1. An inadvertent SIS signal (Train A) is received.
2. Train "B" SIS, which is functioning normally, does NOT actuate.
3. The crew fails to recognize that only the "A" Train of SIS has actuated, and does NOT actuate train "B" SIS.
4. The crew transitions to ES-1.1, *SI Termination*.
5. The RO resets SIS.

At this point, a LOCA occurs, and Containment pressure increases to 19 psia.

Assuming both Trains of SSPS function properly, how will SSPS respond?

a) Neither train of SIS will actuate.
b) The "A" Train of SIS will actuate, but "B" Train will NOT actuate.
c) The "A" Train of SIS will NOT actuate, but "B" Train will actuate.
d) Both trains of SIS will actuate.

Proposed Answer: C

Explanation (Optional): This question includes a lesson learned from the Salem eelgrass event."C" is correct, since "A" Train SIS will not actuate, since the P-4 will block AUTO-SIS after it has been reset ("D" wrong); and "B" Train will actuate, since it hasn't yet actuated, so it hasn't been reset ("A" and "B" wrong). "A" is plausible, since P-4 has actuated, and this would be true if both trains had originally actuated. "B" is plausible, since the two trains are not in the same alignment. "D" is plausible, since this is a misapplication of the effect of SIS Block (versus Reset) switches. HI-1 CTMT pressure SIS is the one input to SIS that is not blocked when operators block SIS when P-11 clears on a controlled RCS cooldown and depressurization.

Technical Reference(s): Functional Sheet 6 (Rev. J)
(Attach if not previously provided, Functional Sheet 7 (Rev. M)
including version/revision number) Functional Sheet 8 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05493 Describe the operation of the following RPS controls and interlocks… P-4… (As available)

Question Source: Bank #80925

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 # 24	Tier #	1	1
K/A Statement: Determine/interpret adherence	Group #	2	2
to Post LOCA Cooldown procedures	K/A #	WE03.EA2.2	
Proposed Question:	Importance Rating	3.5	4.1

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

- Both RHR pumps have just been stopped.
- RCS pressure: 600 psia and stable.
- PZR Level: Empty.
- CTMT Pressure: 20 psia
- CTMT Temperature: 175°F
- CTMT Rad levels: 10^4 R/hr

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

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

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

a) 0900
b) 0908
c) 0916
d) 0924

Proposed Answer: B

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

Technical Reference(s): (Attach if not previously provided, including version/revision number)	E-0 (Rev. 32-0), Note prior to step 1 ES-1.2 (Rev. 20-0), step 5, including Caution prior to step 5

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-07424 Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of ES-1.2.	(As available)

Question Source:	Bank #63963
Question History:	Millstone 3 2013 NRC Exam
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.10
Comments:	

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 25	Tier #	1	1
K/A Statement: Reasons for procedures	Group #	2	2
associated with Steam Generator Overpressure	K/A #	WE13.EK3.2	
Proposed Question:	Importance Rating	2.9	3.3

A turbine runback occurs, resulting in the following sequence of events:

1. The reactor automatically trips.
2. The crew enters FR-H.2, *Response to Steam Generator Overpressure*.
3. FR-H.2 directs the crew to check RCS hot leg WR temperature $\geq$ 530°F.
4. Based on RCS hot leg WR temperature being $\geq$ 530°F, FR-H.2 directs the crew to conduct a cooldown by dumping steam from the unaffected SGs.

Why does FR-H.2 direct the crew to cool down the RCS from the unaffected SGs?

a) To reduce affected SG pressure, since excessive heat transfer from the primary may be the cause of the overpressure condition.
b) To reduce thermal stresses on the U-tubes of the affected SG, since excessive pressure stress already exists on the tubes.
c) To prevent a rapid depressurization of the RCS, since the affected SG may NOT be at saturation conditions.
d) To prevent a radiation release, since the cause of the overpressure condition may be a SG Tube Rupture.

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since excessive heat transfer from the primary to the SG may be the cause of the affected SG overpressurization. Therefore, a check on RCS hot leg temperatures is made. If RCS hot leg temperatures are greater than or equal to 530°F (Which is the saturation temperature of the lowest SG Safety Valve setpoint, including allowances for channel accuracy), a cooldown is initiated by dumping steam from the unaffected SG(s) to aid in reducing the temperature and pressure in the affected SG(s). "B" is plausible, since an overpressure condition exists. "C" is plausible, since this is a basis related to high pressurizer level in FR-I.2, Response to High Pressurizer Level, and the SG may have a high pressure due to overfilling (but this would have led to an earlier transition to FR-H.3, Response to Steam Generator High Level). "D" is plausible, since this is a basis for cooling down the RCS if tube leakage is excessive.

Technical Reference(s): FR-H.2 (Rev. 09-0), step 7
(Attach if not previously provided, WOG Bkgd Doc (Rev. 3) for FR-H.2, step 7
including version/revision number)

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

Learning Objective: MC-05976 Discuss the basis of major procedure steps and/or sequence of steps in EOP FR-H.2. (As available)

Question Source: Bank #86740

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 26	Tier #	1	1
K/A Statement: Operational implications of components, capacity, and function of emergency systems during a loss of Containment Integrity	Group #	2	2
	K/A #	WE14.EK1.1	
	Importance Rating	3.3	3.6

Proposed Question:

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

1. Safety Injection actuates due to four faulted SGs inside Containment.
2. The BOP operator throttles total AFW flow to 550 gpm.
3. All SG NR levels are offscale low.
4. All RCS hot leg temperatures have stabilized as the faulted SGs have depressurized.
5. The crew enters FR-Z.1, *Response to High Containment Pressure.*

The crew is currently checking if Auxiliary Feedwater Flow should continue to all SGs.

Per the Cautions in FR-Z.1, is the crew allowed to reduce AFW flow? If so, to what minimum value? Why or why not?

a) Yes. The crew is allowed to reduce AFW flow down to a minimum of 0 gpm to each SG to minimize the mass/energy release to CTMT.
b) Yes. The crew is allowed to reduce AFW flow down to a minimum of 100 gpm to each SG to minimize the mass/energy release to CTMT.
c) Yes. The crew is allowed to reduce AFW flow down to a minimum total of 530 gpm to ensure minimum heat sink requirements are met.
d) No. The crew is required to increase current AFW flow rates to stabilize RCS temperature.

Proposed Answer: B

Explanation (Optional): The caution prior to step 10 states "If all SGs are faulted, at least 100 gpm feed flow should be maintained to each SG ("B" correct, "A", "C", and "D" wrong)." "A" is plausible since, per step 10 operators are required to isolate AFW flow to a faulted SG (but not with all four SGs being faulted). Per the caution, and the basis document, AFW is required to be throttled to 100 gpm to minimize the energy input to CTMT. "C" is plausible, since in other places in the EOP network, AFW flow is controlled to maintain hot leg temperature, and the cooldown has stopped. "D" is plausible, since normally in the EOPS, AFW flow is maintained at greater than 530 gpm due to heat sink concerns.

Technical Reference(s): FR-Z.1 (Rev. 17-0), step 10, and Cautions prior to step 10
(Attach if not previously provided, WOG Bkgd Doc for FR-Z.1 (Rev. 3), step 6, and Caution
including version/revision number)

Proposed references to be provided to applicants during examination:

Learning Objective: MC-07464 Given a set of plant conditions, properly apply the notes and cautions of FR-Z.1. (As available)

Question Source: Bank #86739

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 27	Tier #	1	1
K/A Statement: Determine/interpret adherence	Group #	2	2
to High Containment Radiation procedures	K/A #	WE16.EA2.2	
Proposed Question:	Importance Rating	3.0	3.3

The reactor has tripped, and the crew has just entered FR-Z.3, *Response to High Containment Radiation Level.*

Which system in addition to the Containment Air Filtration (CAF) System will FR-Z.3 direct the crew to consider using, with ADTS concurrence?

a) Containment Purge System.
b) Containment Vacuum System.
c) Containment Air Recirculation System.
d) Containment Spray System.

Proposed Answer: D

Explanation (Optional): FR Z.3 directs the crew to consider the use of CAF and CTMT spray pumps by discussing these options with the ADTS ("D" correct, "A", "B", and "C" wrong). "A", "B", and "C" are plausible, since each of these are CTMT ventilation systems. Additionally, the Containment Purge can be aligned using the Vacuum System during post accident conditions to reduce Containment Hydrogen concentration (reference E-1). In this alignment, an alternate alignment is made cross-tying these systems to supply a "clean" air supply to containment while aligned for a monitored release path.

Technical Reference(s): FR-Z.3 (Rev. 05-0), step 1
(Attach if not previously provided, E-1 (Rev. 01-0), step 18
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05972 Describe the major action categories within EOP 35 FR-Z.3. (As available)

Question Source: Bank #75470

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 41.8 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 28	Tier #	2	2
K/A Statement: Predict/monitor changes in RCS temperature	Group #	1	1
and pressure associated with operating controls	K/A #	003.A1.7	
Proposed Question:	Importance Rating	3.4	3.4

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

- The Pressurizer is solid
- COPPs is blocked
- RCS is 280 psia
- SG secondary side temperatures are 250 ^{O}F
- No RCPs are running
- Both trains of RHR are aligned for cooldown
- The "B" RHR pump is running

What action/failure would cause an RCS overpressure transient?

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

Proposed Answer: B

Explanation (Optional): "B" is correct since starting an RCP with the PZR solid and SGs greater than 50°F above RCS temperature will cause a pressure increase. This is due to increasing heat transfer from the SGs to the RCS. This heatup could damage the RHR suction line relief valve bellows. "A" is wrong, since starting the 2nd RHR pump would cause the RCS pressure to decrease slightly due to increased heat removal in the RHR heat exchanger, and PCV 131 will reposition in auto to maintain RCS pressure. "A" is plausible, since starting another RHR Pump affects heat removal from the RCS. "C" is wrong since COPPs is blocked. "C" is plausible, since RCS wide range pressure inputs to COPPS.
"D" is wrong since the letdown pressure transmitter failing high will cause the letdown pressure control valve to stroke open to attempt to lower pressure. The valve stroking open will cause RCS pressure to decrease. "D" is plausible, since the repositioning of this valve has a significant effect on RCS pressure while the Pzr is solid.

Technical Reference(s): OP 3201 (Rev. 28-0), Precaution 3.3.4.b
(Attach if not previously provided, Tech Spec LCO 3.4.1.4.1***b (Amendment 230)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07490 Given a set of plant conditions, properly apply the notes and cautions of OP 3201. (As available)

Question Source: Bank #73211

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 # 29	Tier #	2	2
K/A Statement: Operational implications of the effects	Group #	1	1
of RCP shutdown on T-ave, including reason for	K/A #	003.K5.3	
unreliability of T-ave in the shutdown loop	Importance Rating	3.1	3.5

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

1. The "C" RCP Hi-Hi vibration annunciator is received on MB4.
2. The crew enters AOP 3554, *RCP Trip or Stopping a RCP at Power.*
3. The RO stops the "C" RCP.
4. The plant does NOT trip during the transient.
5. The initial transient is stabilized, and SG levels have been restored to 50%.

Assuming the reactor does NOT trip, complete the following statement as to how "C" RCS Loop Tave is affected.

"C" Loop Tave will go to____(1)____of the other three loops, due to____(2)____.

	(1)	(2)
a)	Tcold	increased steam pressure in the "A", "B", and "D" SGs
b)	Thot	increased steam pressure in the "C" SG
c)	Tcold	"C" loop cold leg water passing through the "C" SG before the core
d)	Thot	"C" loop cold leg water passing through the core before the "C" SG

Proposed Answer: C

Explanation (Optional): When the "C" RCP stops, forced circulation stops in that loop. "C" loop driving head is now the DP across the vessel. The running RCPs keep vessel downcomer water (vessel inlet) pressure high, since it is located at their discharge. Flow reverses in the idle loop, since there is a pressure drop across the core. So Tcold water starts reverse-flowing into the "C" loop. The steaming rate in the "C" steam generator will decrease due to less energy being added to it [Q = UA (TAVE -- TSTM)]. So its DT is low, and the entire loop approaches Tcold of the other loops ("C" correct, "B" and "D" wrong). "A" is wrong, since steam pressure decreases in the unaffected loops, since their steaming rate increases to compensate for the loss of steaming from the affected loop. "A" is plausible, since Tave in the affected loop goes to Tcold, and steam pressure is affected in all SGs. "B" is plausible, since Tave is affected, and steam flow from the affected loop decreases significantly. "D" is plausible, since on a loss of all forced flow, Thot increases significantly in all loops while natural circulation flow is being established, and flow will go from the vessel downcomer up through the core before exiting through the hot legs to the SG..

Technical Reference(s): AOP 3554 (Rev. 10-0), step 6
(Attach if not previously provided, AOP 3554 Basis Doc (Rev. 10-0), steps 5 and 6
including version/revision number) P&ID 102A (Rev. 33)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04893 Describe the major parameter changes associated with decreased RCS flow rate. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.3, 41.5 and 41.14

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 30	Tier #	2	2
K/A Statement: Operational implications of the purpose of the	Group #	1	1
CVCS System flowpath around the boric acid storage tank	K/A #	004.K5.31	
Proposed Question:	Importance Rating	3.0	3.4

The plant is at 100% power.

The crew enters AOP 3566, *Immediate Boration.*

Complete the following statement concerning RWST to Charging Pump Suction Isolation Valves 3CHS*LCV112D and 112E while performing this procedure.

These valves are required to be repositioned if__(1)__, and the minimum required flow when aligned through this path is__(2)__.

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

Proposed Answer: D

Explanation (Optional): Step 1 of AOP 3566 attempts to align a boration path from the Boric Acid Tanks (BAT) through the Emergency Boration Valve. If unsuccessful, the crew will attempt to align a path from the BAT through the gravity feed boration path. If the crew is unsuccessful, they will align the path that bypasses the BAT Tanks from the RWST to the RCS. After aligning a boration path, the crew will check minimum boration flow of 33 gpm from the BAT Tanks ("A" and "C" plausible), and if adequate flow does not exist they will align the path from the RWST, with a minimum flow requirement of 100 gpm, since RWST boron concentration is less than that of the BAT Tanks ("D" correct, "C" wrong). "A" and "B" are wrong, since on a reactor trip, immediate boration can continue through its normal path. "A" and "B" are plausible, since if SIS actuates, the crew is required to terminate the immediate boration, since 3CHS*LCV112D and 112E automatically open on the SIS, and having both the RWST and BAT Tanks aligned to the suction of the Charging Pumps at the same time is an unanalyzed condition.

Technical Reference(s): AOP 3566 (Rev. 13-0), Step 3.d.RNO.d.1 and d.4
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03961Describe the major action categories contained within AOP 3566, Immediate Boration. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 31	Tier #	2	2
K/A Statement: Design feature/interlocks for	Group #	1	1
RHR System Modes of operation	K/A #	005.K4.2	
Proposed Question:	Importance Rating	3.2	3.5

Initial conditions:

- The plant is in MODE 4.
- The crew is establishing RHR Train A Boron Concentration in accordance with OP3310A, *Residual Heat Removal System*, and aligning train A of RHR for the "Cooldown" mode.

The RO selects "Cooldown" on the RHR Mode Switch on Main Board 2.

Complete the following statement.

Selecting "Cooldown" on the RHR Mode Switch on MB2 __(1)__ instrument air for RHR Heat Exchanger Outlet Flow Control Valve 3RHS*HCV606 and __(2)__ instrument air for RHR Heat Exchanger Bypass Valve 3RHS*FCV618.

	(1)	(2)
a)	removes	removes
b)	aligns	aligns
c)	aligns	removes
d)	removes	aligns

Proposed Answer: B

Explanation (Optional): When the Normal/Cooldown switch is taken to "Cooldown", air is restored to both valves to allow controlling the cooldown rate ("B" correct, "A", "B", and "C" wrong). Selecting "Normal" removes air from the valves, failing them to their OPEN position, which is an acceptable lineup for SIS or Safety Grade Cold Shutdown cooldown on loss of air. "A". "C", and "D" are plausible, since operating this switch restores or isolates Instrument Air to both of these valves.

Technical Reference(s): OP 3310A (Rev. 18-0), Section 4.5, especially steps 4.5.13, 14, and 15

(Attach if not previously provided, including version/revision number) P&ID 112A (Rev. 50)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05455 Describe operation of the following residual heat removal (RHR) system equipment controls and interlocks… RHR heat exchanger bypass flow control valves…RHR heat exchanger flow control valves… (As available)

Question Source: Bank #72446

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 # 32	Tier #	2	2
K/A Statement: Power supplies to the ECCS	Group #	1	1
valve operators for accumulators	K/A #	006.K2.2	
Proposed Question:	Importance Rating	2.5	2.9

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

- RCS temperature is 450°F
- Pressurizer pressure is 1250 psia
- The SI Accumulator Outlet Valves (3SIL*MV8808A, B, C, and D) are OPEN

How will 3SIL*MV8808A respond if its switch is taken to CLOSE at MB2?

a) The valve will remain open, because power is not available to the MOV.
b) The valve will remain open, due to an RCS pressure interlock.
c) The valve will close and its ESF Status Panel light will illuminate.
d) The valve will close and MB2A Accumulator "A" alarms will NOT illuminate.

Proposed Answer: A

Explanation (Optional): During a plant heatup, when RCS pressure is between 700 psia and 1,000 psia, the accumulator outlet valve MCC breakers will be closed, the valves will be opened from MB2, and then the MCC Breakers will be opened and locked. During normal operation, the Accumulator Outlet Valve MCC Breakers are kept locked open. "A" is correct, and "C" and "D" wrong, since during the cooldown, when RCS pressure is reduced less than 1,000 psia, power is restored to the valves, the valves are closed, and then power is removed from the valves. "C" and "D" are plausible, since when the MCC breaker is closed per procedure, and the MB2 switch is placed in "Close", the valve strokes closed. "B" is wrong, but plausible, since above P-11 (2,000 psia), the accumulator outlet valves receive an auto-open signal. This signal will normally not have an effect on the valve, since it is deenergized.

Technical Reference(s): OP 3310B-4 (Rev 05-0), page 2 of 3
(Attach if not previously provided, OP 3208 (Rev. 30-0), step 4.2.13
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06289 Given a failure (partial or complete) of the Emergency Core Cooling System, determine the effects on the system and on interrelated systems (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 33	Tier #	2	2
K/A Statement: Design feature/interlock	Group #	1	1
for PRT cooling	K/A #	007.K4.1	
Proposed Question:	Importance Rating	2.6	2.9

How does Pressurizer Relief Tank (PRT) Drains Outlet Valve 3DGS-AV8031 respond to a changing PRT level, and why?

a) Automatically closes on PRT high level to prevent water discharge into the Gaseous Waste System vent header.
b) Automatically closes on PRT low level, to prevent lowering PRT level below the minimum level required to quench and cool influent from the Pressurizer.
c) Automatically opens on PRT high level to prevent overfilling the PRT, keeping water out of the Pressurizer Safety Valve discharge piping.
d) Automatically opens on PRT low level to allow lowering level below normal band to allow cooling of the PRT after an extended discharge from the Pressurizer to the PRT.

Proposed Answer: B

Explanation (Optional): Considered a KA match since the interlocks associated with the drain valve ensure adequate cooling water exists for a discharge to the PRT, and manual features that allow for draining and filling the PRT to allow manually cooling the tank. 3RCS-AV8031, PRT Drains Outlet Valve functions to prevent lowering PRT level below the minimum (56%) required for the performance of its design basis function of quenching and cooling influent from the Pzr PORVs and Safety Valves. Interlocks prevent opening the valve until PRT level is raised to 82% ("A" wrong), and will cause the valve to automatically close when PRT level lowers to 62% ("C" and "D" wrong). The Operator must take manual action to open the valve when level increases to 82%, and to override the level interlocks if it is desired to drain the PRT below 62% ("B" correct).

Technical Reference(s):	FSAR Section 5.4.11.1, page 5.4-45 (Rev 27.2)
(Attach if not previously provided,	P&ID 102F (Rev 17-0)
including version/revision number)	P&ID 107A (Rev 27-0)
	LSK-32-3A (Rev 7-0)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-05345 Describe the function and location of the following Pressurizer Relief Tank System components… Pressurizer Relief Tank Outlet Valve DGS-AV8031…	(As available)

Question Source:	Bank #64923
Question History:	Millstone 3 2015 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	55.41.3 and 41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 34	Tier #	2	2
K/A Statement: Ability to perform specific	Group #	1	1
system and integrated plant procedures during	K/A #	008.GEN.2.1.23	
all modes of operation for the CCW system	Importance Rating	4.3	4.4

Proposed Question:

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

- The crew is preparing to place RHR Train "A" in service for plant cooldown.
- The crew desires to use the maximum allowed RPCCW flowrate to the "A" RHR Heat Exchanger to support the cooldown.
- The RO reports the following initial RPCCW flowrates exist:
  - "A" Train Non-Safety Header Flow 1000 gpm
  - "A" Train Safety Header Flow 1700 gpm
  - "A" Train Containment Header Flow 1100 gpm

The RO commences throttling open 3CCP-HK66A1, "RPCCW HX FLOW".

What will the RPCCW flow through "A" RHR heat exchanger indicate when the RPCCW heat exchanger flow limit is reached?

a) 7000 gpm
b) 4300 gpm
c) 4000 gpm
d) 3200 gpm

Proposed Answer: B

Explanation (Optional): The RPCCW Heat Exchanger flow limit is 8,100 gpm, to avoid RPCCW Heat Exchanger tube vibration. 3800 gpm of the maximum 8100 gpm initially exists. Operator can increase flow by 4300 gpm ("C" correct, "A", "B", and "D" wrong). "A" is plausible, since a 7,000 gpm flow limit exists for the RHR Heat Exchanger, but the 8,100 gpm flow limit will be reached prior to reaching this flow. "C" is plausible, since 4,000 gpm is the normal RHR flow rate through the RHR Heat Exchanger. "D" is plausible, since this would be the remaining flow available if the limit was 7,000 gpm, which is the RPCCW flow limit through the RHR Heat Exchanger.

Technical Reference(s): OP 3208 (Rev. 27-0), step 4.3.10
(Attach if not previously provided, OP 3330A (Rev. 23-0), Precaution 3.2
including version/revision number) P&ID 112A (Rev. 50)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04152 Describe the major administrative or procedural precautions and limitations placed on the operation of the Reactor Plant Component Cooling System, and the basis for each. (As available)

Question Source: Bank #69065

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 35	Tier #	2	2
K/A Statement: Ability to locate CCW system	Group #	1	1
control room switches, controls and indications and to	K/A #	008.GEN.2.1.31	
determine they correctly reflect the desired plant lineup	Importance Rating	4.6	4.3

Proposed Question:

During a Main Board walk down, the RO observes the following conditions on MB1:

1. The "C" RPCCW Pump ("A" Train) control switch is normal after stop with its green light lit
2. The "C" RPCCW Pump ("B" Train) control switch is pull to lock

What, if any, actions are needed to restore a normal standby alignment?

A. Nothing needs to be done at the switchgear. The "C" RPCCW pump breaker is normally aligned to the "A" Train with its associated control switch left in the normal after stop.
B. Nothing needs to be done at the switchgear. The "C" RPCCW pump breaker is normally aligned to the "A" Train; however, its control switch needs to be placed in pull to lock.
C. The "C" RPCCW Pump breaker needs to be racked down from its "A" train cubicle, and racked up into its "B" train cubicle. Both control switches ("A" and "B" train) should be in pull to lock.
D. The "C" RPCCW Pump breaker needs to be racked down from its "A" train cubicle, and racked up into its "B" train cubicle. The "A" train control switch should be in pull to lock and the "B" train control switch should be placed in normal after stop.

Proposed Answer: C

Explanation (Optional): The "C" swing pump has one breaker that can be racked up in either train. Normally, the breaker is racked up into the 'B' train cubicle ("A" and "B" wrong), so the crew needs to move the breaker to the "B" train cubicle ("C" correct). Additionally, several years ago the desired plant line-up is to leave both Train's control switches in pull to lock ("C" correct, "A" and "D" wrong). "A" and "B" are plausible for "A" Train alignment as losing cooling on this train causes more time sensitive actions (to isolate CHS and Letdown & subsequently restore RCS inventory control). "A" and "D" are plausible for control switch position in auto as this was the historical position and it's normal to leave racked up components in auto after stop.

Technical Reference(s): OP 3330A (Rev 25-0), 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-04154 Describe the operation of the Reactor Plant Component Cooling System under the following normal, abnormal, or emergency conditions: A. Normal, at power operations B. Shifting Pumps and Heat Exchangers… (As available)

Question Source: Modified Bank #71204

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:
MODIFIED QUESTION BELOW
Bank #71204

The plant is at 100% power, with the "C" RPCCW pump breaker in its normal standby alignment, when the following sequence of events occurs:

1. The "A" RPCCW Pump trips.
2. The crew enters AOP 3561, *Loss of Reactor Plant Component Cooling Water.*
3. As part of the recovery actions, a PEO is dispatched to align the "C" RPCCW pump and heat exchanger to the "A" train.

What local action, if any, needs to be taken at the switchgear?

A. Nothing needs to be done at the switchgear. The "C" RPCCW pump breaker is normally aligned to the "A" train and is already racked up in cubicle 34C10-2.
B. The "C" RPCCW Pump "B" train breaker needs to be racked down, and the "C" RPCCW Pump "A" Train breaker needs to be racked up, since both Trains' breakers are normally installed in their "C" pump breaker cubicles.
C. The "A" RPCCW Pump breaker needs to be racked down from breaker cubicle (34C9-2), and racked up into the "C" RPCCW Pump breaker cubicle (34C10-2).
D. The "C" RPCCW Pump breaker needs to be racked down from its "B" train cubicle (34D9-2), and racked up into its "A" train cubicle (34C10-2).

Proposed Answer: D

Explanation (Optional): The "C" swing pump has one breaker ("C" wrong) that can be racked up in either train ("B" wrong). Normally, the breaker is racked up into the 'B' train cubicle ("A" wrong), so the crew needs to move the breaker to the "A" train cubicle ("D" correct). "A", "B", and "C" are plausible, since two breaker cubicles exist, and the CHS swing pump does not have its own dedicated breaker. It utilizes the associated train's CHS Pump Breaker.

* Question is considered modified as it tests control switch position in addition to breaker position.

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 36	Tier #	2	2
K/A Statement: Power supplies to	Group #	1	1
Pressurizer Heaters	K/A #	010.K2.1	
Proposed Question:	Importance Rating	3.0	3.4

Which Load Center provides power for Pressurizer Backup Heater Group A?

a) 32A
b) 32P
c) 32S
d) 32V

Proposed Answer: C

Explanation (Optional): "C" is correct, and "A", "B", and "D" wrong, since Pzr Backup Heater Group A is powered from Load Center 32S. "A", "B", and "D" are plausible, since these are Load Centers at Millstone 3. Load Center 32A and P provide power to Turbine Building loads. 32V provides power to "B" Train Pzr Heaters.

Technical Reference(s): OP 3301G-001 (Rev. 05-3), page 5 of 5.
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03320 Describe the function and location of the following major 480 volt AC system components… 480 volt Load Centers… (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory of Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 37	Tier #	2	2
K/A Statement: Predict/monitor changes associated with	Group #	1	1
adjusting Reactor Protection System trip setpoints	K/A #	012.A1.1	
Proposed Question:	Importance Rating	2.9	3.4

The Reactor is initially stable at 16% power with all systems in their normal alignment.

Intermediate Range Channel IR-35 fails high.

Assuming no other actions are taken related to IR-35, which of the following correctly states the effect of this failure on automatic power-related reactor trips?

a) The Reactor will trip at the point when power is reduced below $5x10^{-11}$ Amps.
b) The Reactor will trip at the point when power is reduced below 9% power.
c) The Reactor will immediately trip at the current power level of 16% power.
d) The Reactor will trip at the point when power is raised above 25% power.

Proposed Answer: B

Explanation (Optional): This question is considered a KA match since at low power, the Power Range High Flux Trip and the Intermediate Range Overcurrent Trip are set at 25% power. As power is increased, these trips are blocked, effectively changing the overpower trips from 25% power to 109% power. P-10 inputs to the RPS circuit to allow manually blocking the IR high flux trip and rod block, and the PR high flux (low setpoint) trip by depressing the BLOCK pushbuttons on MB4. This is done per OP 3203 when power increases above P-10 (11% power). P-10 also provides a backup input to the SR Reactor trip block, to ensure the SR channels remain de-energized at power. "B" is correct, and "A" wrong, since when P-10 clears at 9% power, the IR High Current trip re-arms, with the IR channel failed high. Also, the IR Trip is a 1 of 2 coincidence trip. "A" is plausible, since P-10 inputs to the P-6 output, and P-6 clears at $5x10^{-11}$ amps, enabling the SR Hi Flux Trip. "C" is wrong, since the IR High Current trip was manually blocked when power was raised above P-10 (11% power). "C" is plausible, since power is initially at 16% power, an IR channel has failed high, and a reactor trip is associated with IR high current. "D" is wrong, since the Power Range High Flux Low Setpoint trip was manually blocked when power was raised above P-10 (11% power). "D" is plausible, since a PR Hi Flux Low setpoint trip occurs at 25% power, and a IR High Current trip occurs at 25% power.

Technical Reference(s): Functional Sheets 3 (Rev. G) and 4 (Rev. G)
(Attach if not previously provided, OP 3203 (Rev. 24-0), step 4.3.9
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…P-10… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 38	Tier #	2	2
K/A Statement: Operational implications of	Group #	1	1
Reactor Protection for DNB	K/A #	012.K5.1	
Proposed Question:	Importance Rating	3.3	3.8

Initial Conditions:

- The plant is operating at 80% power.
- Rod control is in MANUAL.

Power range NIS channel 1 UPPER DETECTOR fails HIGH.

Complete the following statement.

The loop 1 OTΔT Reactor Trip **setpoint** __(1)__ and the loop 1 OPΔT Reactor Trip **setpoint** __(2)__.

	(1)	(2)
a)	decreases	decreases
b)	decreases	unchanged
c)	unchanged	decreases
d)	unchanged	unchanged

Proposed Answer: B

Explanation (Optional): The OTDT Reactor Trip provides primary protection against DNB. If NIS upper detector fails high at 80% power, AFD (Delta I) will become much more positive. Tech Spec/TRM section 2 gives the bases and equations for OTDT and OPDT. For OTDT, the term "f1(DI)" increases on the instrument failure, which subtracts from the OTDT setpoint - indicating increased heat flux in certain areas of the core. OTDT setpoint is reduced by > 1.98% of its value at RATED THERMAL POWER for each % that qt - qb exceeds +3% ("C" and "D" wrong). OPDT does not have an AFD term ("B" correct, "A" wrong). "A", "C", and "D" are plausible, since the terms either decrease or remain the same. Also, it is not obvious as to whether a DNB trip and/or a Kw/foot trip would have an input from AFD.

Technical Reference(s): (Attach if not previously provided, including version/revision number)
Tech Spec Section 2.0 Bases (LBDCR No. 07-MP3-017), page B 2-5
Tech Spec Section 2.0 Bases (LBDCR No. 07-MP3-014), page B 2-6
Tech Spec Table 2.2-1 (Amendment 218), pages 2-9 and 2-10
Tech Spec Table 2.2-1 (Amendment 242), page 2-11
TRM Section 2.0 Bases (LBDCR 16-MP3-011), Sections 2.2.1 and 2.2.2

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #77684

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 # 39	Tier #	2	2
K/A Statement: Power supplies to	Group #	1	1
ESFAS System equipment control	K/A #	013.K2.1	
Proposed Question:	Importance Rating	3.6	3.8

What provides the source of power to the Reactor Protection Train A 48 Volt and 15 Volt Power Supplies?

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

Proposed Answer: A

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

Technical Reference(s): AOP 3564 (Rev. 11-0), Entry Conditions
(Attach if not previously provided, SSPS Power Distribution RPS Training Figure 5 (Rev 05-0)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05245 Discuss the basic power supply arrangement for control and protection channels including the specific channel's color scheme. (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 # 40	Tier #	2	2
K/A Statement: Effect of loss/malfunction of ESFAS	Group #	1	1
sensors and detectors on the ESFAS System	K/A #	013.K6.1	
Proposed Question:	Importance Rating	2.7	3.1

The plant is at 100% power with Pressurizer level control selected to 459/461 (I / III).

A small reference leg leak occurs in pressurizer level transmitter 3RCS*LT459.

Assuming no operator action, what <u>sequence</u> of events will occur?

a.) Letdown isolation occurs.
3CHS*FCV121 throttles open.
Actual pressurizer level starts to increase.
The reactor will eventually trip on high pressurizer level.

b.) 3CHS*FCV121 throttles closed.
Letdown isolation occurs.
Actual pressurizer level starts to increase.
The reactor will eventually trip on high pressurizer level.

c.) Letdown isolation occurs.
3CHS*FCV121 throttles closed.
Pressurizer level will continue to decrease.
The reactor will eventually trip on low pressurizer pressure.

d.) 3CHS*FCV121 throttles closed.
Letdown isolation occurs.
Pressurizer level will continue to decrease.
The reactor will eventually trip on low pressurizer pressure.

Proposed Answer: B

Explanation (Optional): Reference leg failure will cause indicated level to read high, causing FCV-121 to ramp closed. Letdown isolation would initially occur if the channel failed low (A & C are incorrect but plausible as examinee needs to correctly determine how instrument fails with reference leg leak). Actual pressurizer level will decrease due to the failed high controlling channel, and the remaining channels will cause letdown isolation. Seal injection will fill pressurizer ("D" wrong) and eventually cause a high level trip (B is correct). "D" is plausible, since seal leakoff flow continues, and on a loss of all AC, Pressurizer level (and pressure) would decrease.

Technical Reference(s): Functional Dwg 11 (Rev. H)
(Attach if not previously provided, P&ID 104A (Rev. 54)
including version/revision number)

Proposed references to be provided to applicants during examination:

Learning Objective: MC-05338 (RO, SRO, STA) Describe the operation of the Pressurizer Pressure and Level Control System Controls and Interlocks (As available)

Question Source: Bank #73111

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10CFR55.41.7 and 41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 41	Tier #	2	2
K/A Statement: Effect of loss of malfunction of the	Group #	1	1
Containment Cooling System on equipment susceptible	K/A #	022.K3.1	
to damage by temperature, humidity, and pressure	Importance Rating	2.9	3.2

Proposed Question:

With the plant in MODE 3, an earthquake occurs, resulting in the following:

- The "A" CAR Fan trips, and the "B" CAR Fan cannot be started.
- An RCS leak initiates inside Containment.

Over time, this results in the following conditions inside Containment:

- CTMT humidity approaches 100% humidity
- CTMT iodine levels have become significantly elevated
- CTMT particulate levels have become significantly elevated

The crew is considering the use of the Containment Air Filtration (CAF) System to lower radiation levels in CTMT.

Complete the following statement concerning the greatest impact of the elevated CTMT humidity levels on the effectiveness of the CAF System.

The__(1)__filters will be less effective at reducing radiation levels from__(2)__in the Containment atmosphere.

	(1)	(2)
a)	HEPA	particulate
b)	HEPA	iodine
c)	charcoal	particulate
d)	charcoal	iodine

Proposed Answer: D

Explanation (Optional): This question is considered a KA match since Containment Cooling experiences a malfunction (the CAR Fan trips, and the backup fan fails to start) , followed by the applicant having to determine which equipment is susceptible to damage due to elevated humidity (the CAF System). The CAF system is most effective at removing iodine in the charcoal filters ("C" wrong) and particulates in the HEPA filters ("B" wrong). "D" is correct, and "A" wrong, since high humidity conditions significantly reduce the effectiveness of the charcoal filters. "A", "B", and "C" are plausible, since the CAF System has both HEPA and charcoal filters, and removes both particulates and iodine.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	FR-Z.3 (Rev. 005), step 1 FSAR 9.4.8.1 (Rev. 30) www.novent.homestead.com/files/carbon.htm

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04261 Describe the major administrative or procedural precautions and limitations placed on the operation of the Containment Ventilation System, and the basis for each. (As available)

Question Source: Bank #86759

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 41.10 and 41.12

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 42	Tier #	2	2
K/A Statement: Manually operate/monitor	Group #	1	1
Containment Spray reset switches	K/A #	026.A4.5	
Proposed Question:	Importance Rating	3.5	3.5

The crew is responding to a small break LOCA per E-1, *Loss of Reactor or Secondary Coolant*, and the current conditions are as follows:

- CDA actuated 14 minutes ago.
- CTMT pressure is 16.5 psia and decreasing.
- The crew is at E-1, step 7, "Check if CTMT Spray Should Be Stopped".
- The US directs the RO to stop Containment Spray using GA-8, *Stopping Containment Spray*.

Per GA-8, the RO is preparing to depress the CDA Reset pushbuttons on Main Board 2 to allow stopping any running Containment Spray Pumps from MB2.

Complete the following statements concerning the effect, if any, depressing the CDA pushbuttons will have on RPCCW Pump control, and on the Containment Depressurization Actuation annunciator.

The pushbuttons __(1)__ at MB1. The Containment Depressurization Actuation annunciator will __(1)__ on MB2.

	(1)	(2)
a)	do NOT input to RPCCW Pump control	remain lit
b)	do NOT input to RPCCW Pump control	clear
c)	allow starting the RPCCW Pumps	remain lit
d)	allow starting the RPCCW Pumps	clear

Proposed Answer: D

Explanation (Optional): The CDA signal starts the QSS Pumps, and starts the RSS Pumps after a time delay. The CCP (RPCCW) pumps automatically trip on a CDA, so CDA needs to be reset to allow restarting these pumps ("A" and "B" wrong). "A" and "B" are plausible, since the pumps used to depressurize Containment on a CDA are the Spray Pumps. "D" is correct, and "C" wrong, since these pushbuttons clear this annunciator. This provides confirmation that the crew has successfully reset CDA. "C" is plausible, since Containment pressure has dropped below the CDA actuation setpoint of 23 psia, but the signal locks in, and needs to be reset from MB2.

Technical Reference(s): (Attach if not previously provided, including version/revision number)
- E-1 (Rev. 26-0), step 7
- GA-8 (Rev. 02-0), Step 2.b, 3, 4, and 6.RNO.3.b
- Functional Sheet 6 (Rev. J)
- Functional Sheet 7 (Rev. M)
- Functional Sheet 8 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05171 Describe operation of the following containment de-pressurization system components controls and interlocks… QSS pumps… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 43	Tier #	2	2
K/A Statement: Predict impact/mitigate impact	Group #	1	1
of increasing Main Steam System steam demand,	K/A #	039.A2.5	
And its relationship to increases in reactor power	Importance Rating	3.3	3.6

Proposed Question:

The plant is operating at 60% power, and the following initial conditions exist:

- Turbine Impulse Pressure Instrument 3MSS*PT505 has failed low.
- The crew has just entered AOP 3571, *Instrument Failure Response*.
- The Steam Dump MODE Selector is still in the TAVG position

The following sequence of events occurs:

1. All condenser steam dump valves unexpectedly fail open.
2. The crew enters AOP 3582, *Excessive Steam Demand.*

Assuming the reactor does NOT trip, complete the following statement.

The instrument failure that caused the steam dumps to open was__(1)__, and AOP 3582 will direct the crew to__(2)__to restore primary plant parameters to normal.

	(1)	(2)
a)	Turbine Impulse Pressure 3MSS-PT-506 failed LOW	move Control Rods
b)	Turbine Impulse Pressure 3MSS-PT-506 failed LOW	adjust Turbine load
c)	RCS Loop 4 Tave failed HIGH	move Control Rods
d)	RCS Loop 4 Tave failed HIGH	adjust Turbine load

Proposed Answer: B

Explanation (Optional): This question includes OE from a Millstone 2 reactivity management event. When steam demand increases (Heat out of the RCS becomes greater than heat in), RCS temperature decreases, adding positive reactivity. This causes reactor power to increase. Power will increase until heat input from the reactor matches heat out via the secondary, stopping the temperature decrease. The result is increased reactor power with decreased RCS temperature. When 3MSS*PT505 initially failed low, Tref dropped less than Tave, but the dumps are not armed. PT506 failing low arms the steam dumps due to a sensed load reject. With Tref already failed low, the steam dumps will open. "C" and "D" are wrong, since the dumps won't arm on a Tave failure. "C" and "D" are plausible, since Tave feeds into steam dump controls, and if the steam dumps were armed, Tave failing high would cause the steam dumps to fail open. "B" is correct, and “A” wrong, since it is never appropriate to withdraw control rods to restore RCS temperature during a secondary plant transient. This has occurred on more than one occasion in the industry. “A” is plausible, since Tave is low, and withdrawing control rods will raise RCS temperature. Also, the question does not state what happened to power or temperature, and the distractors do not state which way the operators will move control rods or turbine load.

Technical Reference(s): AOP 3582 (Rev. 00-0), Entry Conditions
(Attach if not previously provided, AOP 3582 (Rev. 00-0), step 2.RNO
including version/revision number) Functional Drawing 10 (Rev. J)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04899 Describe the major parameter changes associated with Reactivity & Power Distribution Anomalies (As available)

Question Source: Modified Bank #73475

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1 and 41.5

Comments:

Question is considered "Modified" since new portion of question has been added, asking procedurally directed action to be taken to mitigate the event.

Original Bank Question #73475

The plant is operating at 100% power, and the following initial conditions exist:

- Turbine Impulse Pressure Instrument 3MSS*PT505 has failed low.
- The crew has just entered AOP 3571, *Instrument Failure Response*.
- The Steam Dump MODE Selector is still in the TAVG position

All condenser steam dump valves unexpectedly fail open.

Which additional failure may have caused the steam dumps to open?

a) RCS Loop 4 Tave input to the load rejection controller failed high.
b) RCS Loop 4 Tave input to the load rejection controller failed low.
c) Turbine Impulse Pressure Transmitter 3MSS-PT-506 failed high.
d) Turbine Impulse Pressure Transmitter 3MSS-PT-506 failed low.

Correct Answer: D

Justification:
"D" is correct, when 3MSS*PT505 failed low, Tref dropped less than Tave, but the dumps are not armed. PT 506 failing low arms the steam dumps. With PT-505 also failed low the steam dumps will open. "A" is wrong because Tave input to load rejection controller is auctioneered high. "B" is wrong because the dumps won't arm on the Tave failure. "C" is wrong because PT 506 failing high will not cause steam dumps to arm. "A", "B", and "C" are plausible, since PT 505, 506, and Tave all feed into steam dump controls.

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 44	Tier #	2	2
K/A Statement: Design feature/interlock for reactor	Group #	1	1
building isolation from the Main Steam System	K/A #	039.K4.7	
Proposed Question:	Importance Rating	3.4	3.7

The crew is performing a plant cooldown in accordance with OP 3208, *Plant Cooldown*, and the following initial conditions exist:

- Tave is 540°F and slowly decreasing.
- PZR pressure is 2110 psia and slowly decreasing.
- S/G pressures are all approximately 950 psig and slowly decreasing.

The following sequence of events occurs:

1. A significant steam leak occurs on the "B" steam line in the Main Steam Valve Building.
2. The SG B STEAMLINE PRESSURE LO annunciator is received on MB5B.
3. The SG B PRESSURE RATE HI annunciator is NOT received.
4. No further operator action has been taken.

What is the current status of Containment Isolation Phase A (CIA) and Main Steam Isolation (MSI)?

a) Neither CIA nor MSI occur.
b) CIA actuates only.
c) MSI actuates only.
d) CIA and MSI both actuate.

Proposed Answer: D

Explanation (Optional): Low Steam Pressure SI and MSI are not procedurally blocked until less than 2000 psia. These signals cannot be blocked unless < P-11 (2000 psia). Above P-11, low steam pressure results in a SIS signal, which generates a CIA signal ("A" and "C" wrong), and a MSI signal ("D" correct, "B" wrong). "A" is plausible, since this would be correct if initial pressure was <1950 psia, since the crew would have blocked Low Steam Pressure SI and MSI per OP 3208, and "STEAM PRESSURE RATE HI" did not come in. "B" is plausible since SIS actuates. "C" is plausible, since, per MB5B, 2-4, high steam rate will actuate MSI with SIS blocked < P-11.

Technical Reference(s): (Attach if not previously provided, including version/revision number)
- OP 3208 (Rev. 27-0), step 4.2.5
- OP 3353.MB5B (Rev. 03-0), 3-4
- Functional Sheet 7 (Rev. M)
- Functional Sheet 8 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05493 Describe the operation of the following RPS controls and interlocks… P-11… (As available)

Question Source: Bank # 76240

Question History: Millstone 3 2002 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 45	Tier #	2	2
K/A Statement: Predict/monitor changes associated	Group #	1	1
with operating Main Feed Pump speed controls	K/A #	059.A1.7	
Proposed Question:	Importance Rating	2.5	2.6

With the plant at 80% power, the following sequence of events occurs:

1 The controlling steam flow channel for the "A" Steam Generator starts to slowly fail low.
2. The crew enters AOP 3581, *Immediate Operator Actions.*
3. The BOP operator places the "A" Feed Regulating Valve controller to MANUAL and throttles the valve back to its original position.
4. The RO reports that all 4 SGs Narrow Range levels are 48% and slowly decreasing.

Which of the choices below is procedurally directed per AOP 3581, and is least likely to overshoot the program level setpoint while increasing Feed Pump speed?

a) Allow the TDMFPs to respond in AUTO.
b) Take the Manual Speed Controllers (3TFC-M1A and B) for both TDMFPs to the RAISE direction as needed.
c) Take the TDMFP Master Speed Controller (3FWS-SK509A) to MANUAL and depress the RAISE pushbutton as needed.
d) Take both TDMFP NUS Controller (3FWS-SK46A and B) toggle switches to the RAISE direction as needed.

Proposed Answer: C

Explanation (Optional): "C" is correct, since the master speed controller will raise TDMFP speeds when both TDMFPs are in auto at a controlled rate, while the NUS controllers provide a rapid responding coarse adjust. Procedures direct the use of the Master Speed Controller in Manual, prior to use of the NUS controllers. "D" is wrong, since the toggle switches only work after manual is selected on the NUS controller. "D" is plausible, since the NUS controllers can be used to raise speed. However, the associated lesson plan states the NUS controller toggle switches are "very sensitive, wherein a momentary signal output results in a more dramatic change in feed pump speed than when using the MSC in auto or manual". "B" is wrong, since the manual speed controllers only control feed pump speed when their speed demand is less than the NUS output, and the Manual Speed Changer is already at the high-speed stop to allow full range of control by the NUS controller. "B" is plausible, since the Manual Speed Changer would control speed if it was the lowest set controller. "A" is wrong since Feed Pump Speed Control receives and input from total steam flow to determine program DP. "A" is plausible, since automatic control is the normally preferred method to control feed pump speed.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	OP 3321 (Rev. 24-0), steps 4.3.37 and 42
	AOP 3571 (Rev. 13-0), Attachment M, step M.3
	AOP 3581 (Rev. 04-0), Attachment B, step B.1
	Functional Sheets 13 (Rev. K) and
	Functional Sheet14 (Rev. K)
	FWS059C (Rev. 7-01) Main Feedwater System Lesson Plan
	LSK 6-1.2E (Rev. 10)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04660 DESCRIBE the operation of the following...Controls & Interlocks... Turbine Driven Main Feed Pump Manual Speed Controllers (TFC-M1A/B)... Turbine Driven Feed Pump Speed Controllers (FWS-SK46A/B)... Turbine Driven Main Feed Pump Master Speed Controller (FWS-SK509A)... (As available)

Question Source: Bank #65628

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 # 46	Tier #	2	2
K/A Statement: Effect of loss/malfunction of	Group #	1	1
controllers and positioners on the AFW System	K/A #	061.K6.1	
Proposed Question:	Importance Rating	2.5	2.8

The plant is at 2% power, and initial conditions are as follows:

- Both Motor Driven Auxiliary Feedwater (MDAFW) Pumps are running, feeding all four SGs.
- All AFW Pump suctions are aligned to the CST.

The following sequence of events occurs:

1. Offsite power is lost.
2. The BOP Operator reports the status of AFW as follows:
   - The "B" Motor Driven AFW Pump (3FWA*P1B) is NOT running, and will NOT start.
   - 3FWA*P1A is running.
   - The TDAFW Pump is running.
   - All three AFW Pump Suction Valves are aligned to the CST.
   - Both discharge cross-tie valves (3FWA*AOV62A and B) are OPEN.

In accordance with the cautions of OP 3322, *Auxiliary Feedwater System*, what concern is there with the discharge cross-tie valves open, and which AFW Pump suction valves can the BOP Operator realign to the DWST from MB5?

a) The TDAFW Pump Trip Throttle Valve (3MSS*MSV5) may trip. The BOP Operator can realign only the MDAFW Pump Suction Valves to the DWST from MB5.
b) The TDAFW Pump Trip Throttle Valve (3MSS*MSV5) may trip. The BOP Operator can realign the MDAFW Pump and TDAFW Pump Suction Valves to the DWST from MB5.
c) The "A" MDAFW Pump may reach runout flow. The BOP Operator can realign only the MDAFW Pump Suction Valves to the DWST from MB5.
d) The "A" MDAFW Pump may reach runout flow. The BOP Operator can realign the MDAFW Pump and TDAFW Pump Suction Valves to the DWST from MB5.

Proposed Answer: C

Explanation (Optional): On a LOP, the positioners for the suction valves should have aligned the suction paths for the MDAFW Pumps to the DWST, and the positioners for the discharge cross tie valves should have realigned to the closed position. Operators are cautioned to prevent runout conditions for the MDAFW Pumps with the discharge cross-tie valves open. "A" and "B" are wrong, since the discharge cross tie valves affect the MDAFW Pumps, not the TDAFW Pump. "A" and "B" are plausible, since numerous industry OE exists involving inadvertent tripping of the TDAFW Pump Steam Supply Trip Valve. Millstone 3 has had difficulties with this valve tripping, and this is a concern if the TDAFW pump discharge throttle valves are throttled too quickly. The cross-tie valves could be a concern if all three pump discharges could be cross-tied. The MDAFW pump suction valves (3FWA*AOV23A/B and 3FWA*AOV61A/B) are normally operated from Main Board 5 ("D" plausible), but the TDAFW Pump suction valves (3FWA*HCV37and 3FWA*V30) require local operation ("C" correct, "D" wrong).

Technical Reference(s):	OP 3322 (Rev. 28-0), Section 4.8, caution prior to step 4.8.4, and step 4.8.7
(Attach if not previously provided,	OP 3322 (Rev. 28-0), Precaution 3.4
including version/revision number)	P&ID 130B (Rev. 49)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04635 Describe the operation of the following Auxiliary Feedwater System component controls & interlocks… Motor & Turbine Driven Auxiliary Feedwater DWST Supply Header Isolation Valves… Motor & Turbine Driven Auxiliary Feedwater Pump Alternate Suction … Auxiliary Feedwater Pump Discharge Cross-connect Valves…	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 47	Tier #	2	2
K/A Statement: Manually operate/monitor:	Group #	1	1
local operation of AC Electrical System breakers	K/A #	062.A4.4	
Proposed Question:	Importance Rating	2.6	2.7

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

1. The "Bus 34A Loss of Cntl Pwr" annunciator is received on Main Board 8A.
2. The BOP operator reports the RED light for Bus 34A NSST Feeder Breaker is NOT lit.
3. A PEO is dispatched with an electrician to Bus 34A to monitor local breaker indications.

Complete the following statement concerning local conditions at the breaker.

The ___(1)___ fuse is blown, and the white Auxiliary Circuit light ___(2)___ lit.

	(1)	(2)
a)	UC	IS
b)	UT	IS
c)	UC	IS NOT
d)	UT	IS NOT

Proposed Answer: D

Explanation (Optional): The UT fuse feeds the Red and Green indicating lights on MB8 ("A" and "C" wrong). If a UT fuse blows, the white light will also be dark, since the UT fuse feeds the UC fuse, which feeds the white light ("D" correct, "B" wrong). "A" and "C" are plausible, since the UC fuse feeds a portion of the control circuit (closing coil and white light), and "B" is plausible, since the UC fuse also feeds the white light.

Technical Reference(s): ESK-5A (Rev. 12)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

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. TC coil and circuit. D. UC fuses and circuit. E. UT fuses and circuit. (As available)

Question Source: Bank #68180

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 48	Tier #	2	2
K/A Statement: Physical connections/cause- effect	Group #	1	1
between the AC Electrical System and the ED/G	K/A #	062.K1.2	
Proposed Question:	Importance Rating	4.1	4.4

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

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

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

The Normal 4160V buses are__(1)___, and the Emergency 4160V buses are powered from the__(2)___.

	(1)	(2)
a)	still energized at the low voltage	Emergency Diesels
b)	still energized at the low voltage	RSSTs
c)	de-energized	Emergency Diesels
d)	de-energized	RSSTs

Proposed Answer: A

Explanation (Optional): "C" and "D" are wrong, but plausible, since sustained UV to lockout the NSST for the 4160V buses is 70%, and since voltages is 80%, the NSST supply breakers will not open. After a brownout exists for 4.5 minutes with no SIS signal present, the 4160V buses will attempt to transfer to the RSSTs. But this feature is interlocked with RSST voltage, and will not occur if RSST voltage is less than 97% ("A" wrong, but plausible). This causes the 34A-34C and 34B-34D tie breakers to open, causing the emergency diesels to automatically start and energize the emergency buses from the EDG ("A" correct).

Technical Reference(s): OP 3353.MB8A (Rev. 06-0), 3-12 and 2-2
(Attach if not previously provided, LSKs 24-3.A (Rev. 8), 3.B (Rev. 8), 3.C (Rev. 8), 3.D, (Rev. 12),
including version/revision number) LSKs 24-3.E (Rev. 8), 3.F (Rev. 8), 3.G (Rev. 8), 3.H (Rev. 8),
LSKs 24-3.J (Rev. 9),and 3.K (Rev. 11)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05023 Describe the 4 kV Distribution System operation under normal, abnormal, and emergency conditions: A. At power operations B. Main Generator trip C. Loss of NSSA D. Loss of RSSA (As available)

Question Source: Bank #68084

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 49	Tier #	2	2
K/A Statement: Physical connections/cause-effect between the DC Electrical System and the AC Electrical System	Group #	1	1
	K/A #	063.K1.2	
Proposed Question:	Importance Rating	2.7	3.2

The plant is initially at 100 % power.

480 VAC MCC 32-2T deenergizes due to a bus differential.

Complete the following statement:

DC Bus 1 loads are now being supplied by__(1)__, and VIAC 1 is now being supplied by__(2)__.

	(1)	(2)
a)	Battery Charger 1	the Alternate AC Source via the Static Switch
b)	Battery Charger 1	DC Bus 1 via Inverter 1
c)	Battery 1	the Alternate AC Source via the Static Switch
d)	Battery 1	DC Bus 1 via Inverter 1

Proposed Answer: D

Explanation (Optional): 480 VAC MCC 32-2T supplies the AC input to Inverter 1. This is the normal supply to VIAC 1, since Inverter 1 rectifier output is at 140 VAC, while DC Bus 1 is normally by the charger at 135 VDC. The charger is backed up by Battery 1, which puts out 125 VDC. 32-2T also supplies the normal and swing chargers to DC Bus 1. When 32-2T is lost, Battery 1will pick up DC Bus 1 loads, since its charger has been lost ("A" and "B" wrong). "A" and "B" are plausible, since this would be true if the charger was supplied by another 480 VAC Bus. Automatic switchover to the alternate AC source occurs if the output from the inverter is lost ("C" plausible), but since Battery Bus 1 now supplies the inverter, DC Bus 1 will supply VIAC 1 via the Inverter ("D" correct, "C" wrong).

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

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03296 Describe the function and location of the following major 120 volt AC system components: A. Uninterruptible power supply components: 1. Rectifier 2. Inverter 3. Static Switch 4. Bypass Line Regulator B. Manual Bypass Switch (As available)

Question Source: Bank #78818

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 50	Tier #	2	2
K/A Statement: Manually operate/monitor local/remote	Group #	1	1
operation of the Emergency Diesel Generator	K/A #	064.A4.1	
Proposed Question:	Importance Rating	4.0	4.3

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

1. A PEO starts the "A" Emergency Diesel (EDG) locally for a surveillance run.
2. The PEO places the local Diesel Mode Control Switch in "REMOTE".
3. The PEO reports abnormal noise coming from the "A" EDG.

Complete the following statement concerning emergency-stopping the "A" EDG in these conditions.

The minimum coincidence required when depressing the emergency stop pushbuttons is __(1)__, and the operating location(s) where these pushbuttons function is/are at __(2)__.

a) 1 of 2 the local control panel or at MB8
b) 1 of 2 MB8 only
c) 2 of 2 the local control panel or at MB8
d) 2 of 2 MB8 only

Proposed Answer: D

Explanation (Optional): The minimum coincidence needed to emergency-trip the EDG is 2 of 2 pushbuttons depressed simultaneously ("A" and "B" wrong). "A" and "B" are plausible, since some pairs of switches or pushbuttons, such as SIS actuation switches, have a 1 of 2 coincidence; and some are train-specific, such as SI Reset pushbuttons. "D" is correct, and "C" wrong, since when in REMOTE, the emergency stop pushbuttons at the local control panel will not trip the EDG, only the pushbuttons at MB8. "C" is plausible, since the EDG was started locally, and having emergency trip pushbuttons not function may appear non-conservative.

Technical Reference(s): OP 3346A (Rev. 35-0), section 4.10
(Attach if not previously provided, LSK 24-9.3J (Rev. 8)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05189 Describe the operation of the following emergency diesel generator system components controls and interlocks… emergency stop relay… (As available)

Question Source: Bank #64191

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 51	Tier #	2	2
K/A Statement: Predict impact/mitigate a	Group #	1	1
Process Radiation Monitor detector failure	K/A #	073.A2.2	
Proposed Question:	Importance Rating	2.7	3.2

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

1. SG Blowdown Radiation Monitor (3SSR*RE08) fails high.
2. Blowdown flow isolates.
3. The SM directs Chemistry Department to obtain grab samples to comply with the REMODCM.
4. Chemistry Department requests that the crew restore blowdown flow as soon as possible since the Chemistry Performance Index (CPI) is 1.09.

Complete the following statement.

The Blowdown System valves that automatically closed as a direct result of the SSR08 failure are the SG Blowdown __(1)__, and the minimum action(s) physically required by the crew to allow the re-opening of the closed valves to re-establish blowdown flow is/are __(2)__.

a) (1) Flow Control Valves (3BDG-HV20A-D)
   (2) remove SSR08 from service and depress the Blowdown Sample Isolation Reset pushbuttons on MB1
b) (1) Flow Control Valves (3BDG-HV20A-D)
   (2) depress the Blowdown Isolation Reset pushbuttons on MB1
c) (1) Containment Isolation Valves (3BDG*CTV22A-D)
   (2) remove SSR08 from service and depress the Blowdown Sample Isolation Reset pushbuttons on MB1
d) (1) Containment Isolation Valves (3BDG*CTV22A-D)
   (2) depress the Blowdown Isolation Reset pushbuttons on MB1

Proposed Answer: D

Explanation (Optional): A Hi Rad signal from SSR08 auto-closes the SG Blowdown CIVs ("A" and "B" wrong). The Hi Rad signal can be overridden by resetting blowdown isolation on MB1, even with the initiating signal still present ("C" wrong, and "D" correct). "A" and "B" are plausible, since 3BDG-HV20A-D also receive AUTO-CLOSE signals, but not from Hi Rads, and 3BDG-HV20A-D will stop blowdown flow if closed. "A" and "C" are plausible, since the Blowdown Sample Isolation Reset pushbuttons are also on MB1, and for some reset circuits, such as SIS reset, the SIS will come right back in after a reset if the actuating signal is still present.

Technical Reference(s): AOP 3573 (Rev. 23-0), Att. A, page 12 of 12.
(Attach if not previously provided, LSK 32-13A (Rev. 9)
including version/revision number) LSK 32-13C (Rev. 4)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05472 Given a failure of the Radiation Monitoring System (partial or complete), determine the effects on the system and on inter-related systems. (As available)

Question Source: Bank #80885

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.11

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 52	Tier #	2	2
K/A Statement: Ability to determine operability	Group #	1	1
and/or availability of safety related equipment	K/A #	076.GEN.2.2.37	
related to the Service Water System	Importance Rating	3.6	4.6

Proposed Question:

The "A" EDG is running, when a mat of debris in the Service Water System severely fouls the "A" EDG Engine Air Cooler Water (Intercoolant) Heat Exchanger.

Regarding the below listed "A" EDG Systems, identify the system or systems, if any, that will be adversely impacted by the Service Water fouling of the Intercoolant Heat Exchanger.

- Jacket Water Cooling System
- Lube Oil System

a) Neither system
b) Only the Jacket Water Cooling System
c) Only the Lube Oil System
d) Both systems

Proposed Answer: D

Explanation (Optional): Service Water flows in series from the Air Cooler Water (Intercoolant) Heat Exchanger to the Jacket Water Heat Exchanger, so Jacket Water temperature would be impacted with increased temperatures ("A and "C" wrong). Since Jacket Water cools the Lube Oil Heat Exchanger, the Lube Oil system will be similarly affected ("D" correct, "B" wrong).

Technical Reference(s): P & ID 116A (Rev. 47)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04395; Describe the function and location of the following EDG (and Support) system components... Jacket Water Cooling system... Intercoolant system... Lube Oil system... (As available)

Question Source: Bank #78428

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 53	Tier #	2	2
K/A Statement: Monitor automatic operation of	Group #	1	1
the Instrument Air System due to air pressure	K/A #	078.A3.1	
Proposed Question:	Importance Rating	3.1	3.2

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

1. A PEO reports an air leak on the instrument air ring header in the turbine building.
2. The RO reports that instrument air header pressure is at 88 psig and decreasing slowly.
3. The RO is observing instrument air pressure for indications of service air to instrument air cross tie valve 3IAS-AOV14 opening and service air supply valve 3SAS-AOV33 closing.

Assuming the added capacity of the Service Air Compressor is capable of compensating for the leak, complete the following statement as to how instrument air header pressure will respond prior to any local actions by PEO's.

After IAS header pressure lowers to___(1)___, pressure will start to recover when 3IAS-AOV14 and 3SAS-AOV33 realign. Pressure will start to drop again when IAS header pressure increases to___(2)___.

	(1)	(2)
a)	80 psig	103 psig, since the valves will realign to their original positions
b)	80 psig	110 psig, since the running instrument air compressors will unload
c)	85 psig	103 psig, since the valves will realign to their original positions
d)	85 psig	110 psig, since the running instrument air compressors will unload

Proposed Answer: C

Explanation (Optional): Pressure switch 3IAS-PS14, which senses IAS common header pressure downstream of the IAS receivers, will cause AOV14 to open, and AOV33 to close when pressure lowers to 85 psig ("A" and "B" wrong). Additionally, when IAS header pressure increases to 103 psig, PS14 will automatically realign the AOV's to their normal positions, and pressure will again start to decrease since the leak has not been isolated ("C" is correct and "D" is wrong). "A" and "B" are plausible, since numerous setpoints exist based on IAS pressure. "D" is plausible, since if the cross tie valve did not realign at 103 psig, the running IAS compressors would unload at 110 psig.

Technical Reference(s): OP 3353.IS (Rev. 03-0), 1-1
(Attach if not previously provided, LSK-12-1C (Rev. 6)
including version/revision number) LSK-12-2C (Rev. 8)
PAS Training Lesson Plan (Rev. 7), slide number 59

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05321 Describe the operation of the following plant air systems components controls and interlocks… Service air to instrument air cross-connect valve (IAS-AOV14)… (As available)

Question Source: Bank #67323

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4 and 41.77

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 54	Tier #	2	2
K/A Statement: Predict impact/mitigate	Group #	1	1
emergency containment entry	K/A #	103.A2.5	
Proposed Question:	Importance Rating	2.9	3.9

With the plant at 100% power, the crew enters AOP 3568, *Emergency Breaking of Containment Vacuum*, in preparation for making an emergency containment entry.

The BOP Operator is reviewing a caution in AOP 3568 that warns the crew NOT to open 3HVU*CTV32A or 3HVU*CTV32B (containment purge isolation valves) during the containment repressurization.

The reason for this caution is that opening either 3HVU*CTV32A or 3HVU*CTV32B risks_____________.

a) radioactive contamination of Auxiliary Building ductwork after pressure equalizes
b) damage to ventilation ductwork in the Auxiliary Building
c) an unmonitored radioactive release to the environment after pressure equalizes
d) damage to ventilation ductwork in Containment

Proposed Answer: B

Explanation (Optional): The reason for the caution concerning keeping the CTVs closed is that pressure gradients between the Aux Building and Containment experienced during depressurization and repressurization of the containment structure require closure of containment purge air valves, 3HVU*CTV32A and 3HVU*CTV32B, to prevent damage to sheet metal ductwork in the Auxiliary Building ("B" correct). "A" is wrong but plausible since the CTMT atmosphere has yet to be tested and could contain small amounts of airborne activity. Opening either valve would provide an opening from CTMT to the Aux Building ventilation ductwork, which is not going to be exposed to the containment atmosphere during repressurization. 'C' is wrong but plausible since opening 3HVU*CTV32A provides a path to the Aux Building roof, and opening 3HVU*CTV32B provides a path to the turbine building stack. 'D' is wrong but plausible since opening 3HVU*CTV33A will expose CTMT ductwork to high pressure and flow during repressurization.

Technical Reference(s): (Attach if not previously provided, including version/revision number)
AOP 3568 (Rev 03-2), Caution prior to step 2
OP 3313F (Rev 13-0), Precaution 3.8

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04250 (215567); Describe operation of the Containment System under the following emergency conditions: Containment entry, loss of CTMT integrity and emergency breaking of CTMT vacuum. (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 55	Tier #	2	2
K/A Statement: Effect of loss of malfunction of	Group #	1	1
Containment integrity under normal operations	K/A #	103.K3.2	
Proposed Question:	Importance Rating	3.8	4.2

A refueling outage has just been completed, and initial conditions are as follows:

- A plant heatup is in progress per OP 3201, *Plant Heatup*.
- The RCS is at 280°F.

The following sequence of events occurs:

1. The RO reports Containment pressure is 13.9 psia, and slowly increasing.
2. The crew finds an open manual vent valve at a containment penetration.
3. The crew determines the valve had been inadvertently left open and closes the valve.
4. The crew verifies normal radiation conditions exist in Containment.
5. The US directs the RO to operate the Containment Vacuum (CVS) System to lower Containment pressure using OP 3313E, *Containment Vacuum*.

Complete the following statement.

Regarding the use of the Containment Air Ejector and/or the Containment Vacuum Pumps, per OP 3313E, the RO should align______to attempt to restore Containment pressure to the desired value.

a) The Containment Air Ejector only
b) One Containment Vacuum Pump only
c) One or both Containment Vacuum Pumps only
d) The Containment Air Ejector and both Containment Vacuum Pumps

Proposed Answer: C

Explanation (Optional): Explanation (Optional): The Ctmt Vacuum Air Ejector is used to initially drawdown Ctmt pressure to its normal operating range, and is much quicker at removing air than the vacuum pumps ("A" and "D" plausible). "A" and "D" are wrong, since the Air Ejector is not allowed to be aligned unless the plant is in MODES 5 or 6, since Tech Spec 3.6.5.1 requires the Ctmt isolation valves for the air ejector suction line to be closed in MODEs 1, 2, 3, and 4. "A" and "D" are also plausible, since the plant is in a lower MODE. "C" is correct, and "B" wrong, since using both Vacuum Pumps at once is allowed. "B" is plausible, since normally, one Vacuum Pump is operated inter-mittently to compensate for air in-leakage after normal operating containment pressure has been reached.

Technical Reference(s): OP 3313E (Rev. 09-0), Section 1.2 and Precautions 3.1 and 3.3
(Attach if not previously provided, OP 3313E (Rev. 09-0), Note prior to step 4.2.1
including version/revision number) Tech Spec LCO 3.6.5.1 (Amendment No. 258)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04270 Describe the major administrative or procedural precautions and limitations placed on the operation of the Containment Vacuum System, and the basis for each. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.9 and 41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 56	Tier #	2	2
K/A Statement: Effect of loss/malfunction	Group #	2	2
of pumps on the Reactor Coolant System	K/A #	002.K6.7	
Proposed Question:	Importance Rating	2.5	2.8

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

1. Offsite power is lost.
2. A four-square inch small break LOCA occurs on the "D" RCS Cold Leg.
3. Safety injection actuates.
4. No Charging Pumps are running, and none can be started.
5. No SIH Pumps are running, and none can be started.
6. Both RHR Pumps are running.

Per WOG accident analysis, what will be the effect on the RCS of the loss of all high-head safety injection pumps?

a) RCS pressure will steadily decrease to accumulator injection pressure. Since break flow is adequate to remove all decay heat, SGs will no longer be removing heat, and accumulator injection will keep the core covered throughout the transient.
b) RCS pressure will steadily decrease to a pressure where mass into the RCS from the accumulators equals mass out of the break. Since break flow is not adequate to remove all decay heat, SGs will continue to remove excess decay heat via reflux boiling while slight core uncovery occurs.
c) RCS pressure will quickly decrease to, and stabilize at, approximately 1250 psia. The upper portions of the core will start to uncover, but the loop seal will clear in the RCS crossover leg, and shortly thereafter level in the fuel will recover, remaining covered until accumulators inject.
d) RCS pressure will quickly decrease to, and stabilize at, approximately 1250 psia. The loop seal will clear in the RCS crossover leg, but deep core uncovery will exist for several minutes before pressure drops low enough for accumulators to start recovering core level.

Proposed Answer: D

Explanation (Optional): "D" is correct, since on a small cold leg break, the vessel will draw a bubble and act as a pressurizer due to decay heat addition. The cold leg will remain full of water, limiting heat removal out the break, so decay heat input exceeds heat removal out the break. Excess heat will be removed through the steam generator reliefs, resulting in RCS pressure stabilizing at a pressure slightly above the steam generator pressure ("A" and "B" wrong). Even with equilibrium RCS pressure, break flow continues depleting RCS inventory. When the loop seal clears, steam will flow through the break, resulting in increased heat removal, so RCS depressurization occurs, but the fuel will remain uncovered until RCS pressure reaches accumulator pressure ("C" wrong). "A" is plausible, since this is the basic response to a hot leg break where break size is large enough to remove all decay heat. "B" is plausible, since this is the basic pressure response expected for a small break LOCA that is small enough that makeup can replace all the water lost out the break, except pressure would stabilize above SG pressure. "C" is plausible, since this is basic trend for a small cold leg break with high head ECCS available.

Technical Reference(s): Westinghouse MITCORE Core Cooling Text (1991), Figures 2.2.4 and 2.2.5

(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04933 Assuming no operator action, analyze the loss of core cooling event leading to core damage (As available)

Question Source: Bank #80924

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5, 41.8, and 41.14

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 57	Tier #	2	2
K/A Statement: Power supplies to NIS channels, components and interconnections	Group #	2	2
	K/A #	015.K2.1	
Proposed Question:	Importance Rating	3.3	3.7

Complete the following statements concerning Control Power for Intermediate Range (IR) NIS Channel N31.

Control Power is provided from___(1)___. Control power provides power to ___(2)___.

	(1)	(2)
a)	VIAC 1	Detector compensating voltage
b)	VIAC 1	the Bistable lamps
c)	Battery 1	Detector compensating voltage
d)	Battery 1	the Bistable lamps

Proposed Answer: B

Explanation (Optional): Control Power is provided from VIAC 1 ("C" and "D" wrong), and provides power to the Bistable lamps on the NIS Power Range Drawers ("B" correct, "A" wrong). "A" is wrong, but plausible, since instrument power from VIAC 1 supplies compensating voltage. "C" and "D" are plausible, since Battery 1 supplies control power to numerous safety related circuits, and this question asks about control power, not instrument power.

Technical Reference(s): Training Drawing NIS-15 (Rev. 5/0) Control Power Drawing (Based on NIS Tech Manual Drawing)
(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… Blown Power Range control power fuse… (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 # 58	Tier #	2	2
K/A Statement: Operational implications of	Group #	2	2
separation of control and protection circuits	K/A #	016.K5.1	
Proposed Question:	Importance Rating	2.7	2.8

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

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

a) There will NOT be any immediate effect on RPS.
b) A fuse will blow in the associated RPS input relay bay.
c) The associated protection system bistable will trip.
d) The associated RPS alarms will actuate, but the bistable will NOT trip.

Proposed Answer: A

Explanation (Optional): "A" is correct, and "B", "C", and "D" wrong, since control signals going to the controller are electrically isolated from the protection circuits, to prevent electrical faults in control circuits from affecting safety circuits. "B" is plausible, since an overcurrent condition could exist in RPS if the control system was not electrically isolated from the protection circuit. "C" is plausible, since the circuits use the same input signals. "D" is plausible, since alarm circuits are upstream of the trip bistables.

Technical Reference(s): Process Sheet 12 (Rev. N)
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05243 Discuss the difference between "Process Control" and "Process Protection". (As available)

Question Source: Bank #68324

Question History: Millstone 3 2013 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 # 59	Tier #	2	2
K/A Statement: Predict/monitor changes in Spent Fuel	Group #	2	2
Pool water level associated with operating Spent Fuel	K/A #	033.A1.1	
Pool Cooling System controls	Importance Rating	2.7	3.3

Proposed Question:

What would cause PGS Makeup Valve to the Spent Fuel Pool (3SFC-LV44) to open?

a) It will open automatically with the Fuel Pool Cooling Panel LV-44 control switch in AUTO <u>when</u> Spent Fuel Pool level drops below 44%.
b) It will open automatically with the Fuel Pool Cooling Panel LV-44 control switch in AUTO <u>when</u> Spent Fuel Pool level drops below 35%.
c) It will open manually <u>only</u> if Spent Fuel Pool level is below 44%, using the OPEN pushbutton on either the Fuel Pool Cooling Panel OR Main Board 1.
d) It will open manually <u>only</u> if Spent Fuel Pool level is below 35%, using the OPEN pushbutton on either the Fuel Pool Cooling Panel OR Main Board 1.

Proposed Answer: C

Explanation (Optional): The normal SFP makeup valve, 3SFC-LV44, is controlled from MB1 or the Fuel Pool Cooling Panel by Open/Auto - Close pushbutton controllers. Makeup valve LV-44 is normally opened manually when SFP level reaches 40% level (Low Level Alarm is at 35%). The makeup valve is normally opened at MB1 but can be opened at the FP. When SFP level reaches 44%, level transmitter LT-26 provides a Close signal for LV-44. Either of the controllers in the Close position will also cause LV-44 to close. "C" is correct, since LV44 will open if the open pushbutton at either the Fuel Pool Cooling Panel or MB1 is depressed, and SF Pool level is less than the auto-close setpoint of 44% ("D" wrong). "D" is plausible, since 35% is the low level alarm setpoint. "A" and B" are wrong, since there is no auto-open feature. "A" and "B" are plausible, since there is an auto-close feature for the valve.

Technical Reference(s): OP 3305 (Rev 25-0), Section 4.13.2.a
OP 3353.MB1A (Rev. 08-0), 3-4
LSK 34-1B (Rev. 9)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05641 Describe the operation of the following Spent Fuel Pool Cooling System controls and interlocks… Spent Fuel Pool Makeup Valve (SFC-LV44) (As available)

Question Source: Bank #65338
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 # 60	Tier #	2	2
K/A Statement: Manually operate/monitor neutron levels in	Group #	2	2
the Control Room associated with Fuel Handling Equipment	K/A #	034.A4.2	
Proposed Question:	Importance Rating	3.5	3.9

The plant is in MODE 6, with current conditions as follows:

- Fuel handling operations are in progress, with new fuel being loaded into the core from the Spent Fuel Pool.
- Both Westinghouse Source Range NIS channels are operating normally.
- Gammametrics Channel 2 is NOT available
- Gammametrics Channel 1 is operating normally
- The following NIS switch positions are selected at the NIS Cabinets:
  - The "Channel" Selector Switch on the Audio Count Rate drawer is in the "N-31" position.
  - The "Storage" Selector Switch on the rear of the Audio Count Rate drawer is in the "Normal" position.
  - The "Audio Multiplier" Switch on the Audio Count Rate drawer is in the "10" position.

What visual indications of neutron counts are available at the NIS cabinets, and what audible indication of counts is heard in the Control Room?

	Visual indication of neutron counts	Audible indication of neutron counts
a)	Westinghouse and Gammametrics	Westinghouse
b)	Westinghouse and Gammametrics	Gammametrics
c)	Westinghouse only	Westinghouse
d)	Westinghouse only	Gammametrics

Proposed Answer: A

Explanation (Optional): Westinghouse and Gammametrics Source Range indications exist in the Control Room at the NIS Cabinets ("C" and "D" wrong), since all of the switches in the question stem only affect audible counts. "C" and "D" are plausible, since the channel selector switch is in series with the Amplifier Selector Switch, only Westinghouse NIS is available at MB4, another Gammametrics indication exists at the Aux Shutdown Panel area, and audible counts for the Control Room and Containment tap off the circuit at different locations. Audible counts is aligned to Westinghouse when in the "Normal" position ("A" correct). "B" is wrong, since Gammametrics Channel "1" is not capable of emitting sound. "B" is plausible, since Gammametrics channel "2" is capable of providing audible indication in the Control Room, but it is deenergized.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	OP 3360 (Rev. 07-9), Section 4.7 Training Drawing NIS-22 (Rev. 5/0) (Based on Tech Manual)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-05221 Describe the function and location (Control Room and Auxiliary Shutdown Panel) of the following major Ex-core and Incore Nuclear Instrumentation System components… Audio Count Rate/Timer – Scaler…	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 61	Tier #	2	2
K/A Statement: Monitor automatic operation	Group #	2	2
of Steam Generator Water Level Control	K/A #	035.A3.1	
Proposed Question:	Importance Rating	4.0	3.9

With the plant initially at 80% power, the following sequence of events occurs:

1. The extra operator is monitoring the Steam Generators at MB5 to verify proper operation of SGWLC.
2. Prior to seeing any effects from boration, the crew commences lowering turbine load at 1% per minute per AOP 3575, *Rapid Downpower.*

Complete the following statement about how the Main Feed Regulating Valves will respond to the lowering turbine load.

The Main Feed Regulating Valves initially throttle in the__(1)__signal. This will be followed by the valve throttling as needed to__(2)__.

	(1)	(2)
a)	closed direction due to the level error	match feed flow and steam flow
b)	open direction due to the level error	match feed flow and steam flow
c)	closed direction due to the flow error	restore SG levels to 50%
d)	open direction due to the flow error	restore SG levels to 50%

Proposed Answer: C

Explanation (Optional): SGWLC receives a "flow error" signal from feed flow versus steam flow, and a "level error" signal, which compares actual Narrow Range level to the level setpoint of 50% ("A", "B", and "D" plausible). As steam flow decreases when the downpower commences, steam flow becomes less than feed flow, and SG inventory increases due to a decrease in steam flow. However, SG level is measured in the downcomer region of the SGs, which experiences "shrink" as the downpower is commenced, creating an artificially low level indication. SGWLC is designed to respond initially to the flow error signal, which anticipates the change in SG inventory, while the level error signal is lagged due to shrink or swell. So initially, the feed control valve throttles in the close direction ("D" wrong) due to flow error ("A" and "B" wrong), and then adjusts as needed to restore level to 50%, since SGWLC is also designed to be level dominant ("C" correct).

Technical Reference(s): Functional Sheet 13 (Rev. K)
(Attach if not previously provided, Functional Sheet 14 (Rev. K)
including version/revision number) Process Block Sheet 33 (Rev. J)

Proposed references to be provided to applicants during examination:

Learning Objective: MC-04660 Describe the operation of the following Main Feedwater & Steam Generator Water Level Control Systems Controls & Interlocks… Main Feed Regulating Valves (FWS*FCV510/520/530/540)… (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 62	Tier #	2	2
K/A Statement: Ability to verify that Waste Gas Disposal	Group #	2	2
System alarms are consistent with plant conditions	K/A #	071.GEN.2.4.46	
Proposed Question:	Importance Rating	4.2	4.2

With the plant at 100% power the following valves automatically reposition in the plant:

- Degasifier Feed Preheater Steam Supply Valve 3ASS-TV31 closes.
- Degasifier Preheater Condensate Divert Valve 3CNA-AOV46 diverts to the Aux Building Sump.

A PEO is dispatched to the Gaseous Waste Panel to investigate.

Which alarm at the Gaseous Waste Panel would have caused the automatic valve responses that occurred?

a) DEGASIFIER FEED PRESSURE LO
b) GAS WASTE FEED PREHTR OUT COND HI
c) PROCESS GAS CHARCOAL BED RADIATION HI
d) PROCESS GAS RECEIVER PRESSURE HI

Proposed Answer: B

Explanation (Optional): Upon receipt of high conductivity, 3CNA-AOV 46 diverts to the Aux Building Sump AND 3ASS-TV31 closes ("B" correct). "A" and "D" are wrong, since these annunciators do not have automatic actions associated with them. "C" is wrong, since this annunciator causes the Process Gas Receiver Pressure Control Valve (3GWS-PV49 to automatically close. "A, "C", and "D" are plausible, since all of these are actual alarms on the GWS panel, and numerous alarms at the GWS Panel (such as degasifier high level) result in automatic actions.

Technical Reference(s): OP 3353.GW (Rev. 01-0) 1-1, 2-2, 2-5, and 2-6
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04733 (315650) Given a failure, partial or complete, of the GWS system, determine the effects on the system and on interrelated systems. (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 # 63	Tier #	2	2
K/A Statement: Area Radiation System design feature/interlock which provides for containment ventilation isolation	Group #	2	2
	K/A #	072.K4.1	
Proposed Question:	Importance Rating	3.3	3.6

The plant is in MODE 5, with the CTMT Purge System (HVU) in operation.

Fuel Drop monitor 3RMS*RE42 goes into HIGH alarm.

What immediate automatic actions, if any, occur on the running CTMT Purge Exhaust Fan (3HVR-FN4A or 4B), and/or on the purge supply valves (3HVU*CTV32A and 33A) and exhaust valves (3HVU*CTV32B and 33B)?

a) The Fan continues to run. One supply valve and one exhaust valve close.
b) The Fan trips. One supply valve and one exhaust valve close.
c) The Fan continues to run. All four supply and exhaust valves close.
d) The Fan trips. All four supply and exhaust valves close.

Proposed Answer: A

Explanation (Optional): High Radiation isolates the CTMT Purge Supply and Exhaust paths if either fuel drop monitor goes into alarm. One monitor isolates the supply and return inside containment valves, and the other monitor isolates the outside containment valves ("C" and "D" wrong). The fans do not receive an auto-trip signal ("A" correct, "B" wrong). "C" and "D" are plausible, since each of these isolates CTMT. "B" is plausible, since the fan is running without a suction path.

Technical Reference(s): (Attach if not previously provided, including version/revision number)
AOP 3573 (Rev. 23-0), Attachment B, page 5 of 5
P&ID 148A (Rev. 42)
P&ID 153A (Rev. 29)
LSK 22-1D (Rev. 7)
LSK 22-27E (Rev. 6)

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: A. RMS-RE-41/42... (As available)

Question Source: Bank #73071

Question History: Millstone 3 2011 NRC Exam

Question Cognitive Level: Memory of Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.11

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 64	Tier #	2	2
K/A Statement: Physical connections/cause-effect between Rod Control and NIS/RPS	Group #	2	2
	K/A #	001.K1.05	
Proposed Question:	Importance Rating	4.5	4.4

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

1. Power Range NI Channel N44 rapidly fails high.
2. Control Bank "D" rods start to drive in.

Assuming no operator action is taken, complete the following statements about how the Rod Control System responds to this event.

Automatic inward rod motion will__(1)__. Prior to switch operation at the NIS cabinet, rods__(2)__be withdrawn manually at MB4.

	(1)	(2)
a)	stop as the power mismatch signal decays	CANNOT
b)	stop as the power mismatch signal decays	CAN
c)	continue until rods are fully inserted	CANNOT
d)	continue until rods are fully inserted	CAN

Proposed Answer: A

Explanation (Optional): The power mismatch signal (difference in the rate of change between reactor and turbine load) will initially cause rods to drive in due to a rapid increase in auctioneered high NIS power. As rods drive in, Tave decreases. Temperature error will start offset the inward rod motion signal, and the and the power mismatch signal will decay, since it is not a primary to secondary power mismatch signal ("C" and "D" wrong), but a rate of change signal. This would result in inward rod motion stopping. The overpower rod stop coincidence is 1 of 4 channels, so outward rod motion is blocked until the failed channel is defeated at the NIS cabinets ("A" correct and "B" wrong). "C" and "D" are plausible, since this would be true if a temperature instrument failed high, since temperature error is based on difference between Tave and Tref, not a rate of change between the two. "B" is plausible, since most of the RPS coincidences are 2 of 4 channels, not 1 of 4.

Technical Reference(s): Functional Sheet 9 (Rev. H)
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05477 Describe the operation of the following Rod Control System controls and interlocks: A. Manual Rod Control Switch B. Bank Selector Switch C. Control Interlocks 1. C-1 2. C-2... (As available)

Question Source: Bank #64983

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 10 CFR 55.41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 65	Tier #	2	2
K/A Statement: Predict impact/mitigate	Group #	2	2
low Fire Protection header pressure	K/A #	086.A2..2	
Proposed Question:	Importance Rating	3.0	3.3

While performing construction work on the west side of the Unit 3 Turbine Building, the following sequence of events occurs:

1. A backhoe inadvertently causes a moderate size rupture in the Fire Protection Water (FPW) header.
2. The FPW System header pressure drops to and stabilizes at 70 psig.

Prior to operator action, what is the status of the FPW pumps?

a) The Diesel Driven Fire Pump will be running only.
b) The Unit 3 Electric Driven Fire Pump will be running only.
c) Both Electric Driven Fire Pumps will be running only.
d) All three of the Main Fire Pumps will be running.

Proposed Answer: D

Explanation (Optional): The first electric pump auto-starts at 95 psig decreasing ("B" plausible), and the second starts at 85 psig decreasing ("C" plausible). Continued decrease in pressure will cause the diesel pump to start at 75 psig decreasing. "D" is correct, and "A", "B", and "C" wrong, since header pressure has dropped to 50 psig, all three pumps are running. "A" is plausible, since on several systems, such as AFW, Lube Oil, and Seal Oil, the non-electric pumps receive different start signals than the electric pumps.

Technical Reference(s): MP3 FPER (Rev. 30-0), Section 4.1.1
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04610 Describe the function and location of the following Water Fire Protection (FPW) system components…
Electric Fire Pumps (2 total)… Diesel Fire Pump… (As available)

Question Source: Bank #67312

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4 and 41.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 66	Tier #	3	3
K/A Statement: Knowledge of administrative	Group #	1	1
requirements of temporary management directives such	K/A #	GEN.2.1,15	
standing orders, night orders, operations as memos, etc.	Importance Rating	2.7	3.4

Proposed Question:

A Temporary Standing Order has just been issued per OP-AA-100, *Conduct of Operations*.

Complete the following statement concerning what is normally allowed / required with Standing Orders.

Per OP-AA-100, Standing Orders_________.

a) can dictate guidance that is inconsistent with approved station procedures provided the procedures are changed within 48 hours
b) do NOT always require Operations Management approval
c) can be issued to allow less restrictive surveillance acceptance criteria with FSRC approval
d) may impose more restrictive Tech Spec requirements when it is determined that Tech Spec requirements are improper or inadequate

Proposed Answer: D

Explanation (Optional): Temporary orders are used by Operation's Management to clarify operational information of a temporary nature. Two types of temporary orders exist: (1) Shift Orders: deal with day-to-day operational issues and (2) Standing Orders: deal with operational requirements of a continuing nature.
"A" is wrong, since temporary orders (Standing or Shift) may not contradict plant procedures. "A" is plausible, since these orders often do provide amplifying or clarifying information.
"B" is wrong, since temporary orders are approved by Operations Management. "B" is plausible, since these orders are temporary in nature.
"C" is wrong, since standing orders are not normally used to change procedures or support operability, and if they are used to support operability, it is only when Tech Specs are determined to be inadequate or improper. "D" is correct, and "C" plausible, since if a Tech Spec is found to be inadequate, a standing order may be issued to impose more restrictive requirements.

Technical Reference(s): OP-AA-100 (Rev. 33-0), Att. 7, page 1 of 2.
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05072 Discuss on-coming shift responsibilities (As available)

Question Source: Bank #369709

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 67	Tier #	3	3
K/A Statement: Knowledge of how to conduct	Group #	1	1
system lineups, such as valves, breakers, switches, etc.	K/A #	GEN.2.1.29	
Proposed Question:	Importance Rating	4.1	4.0

An operator has completed the initial positioning (valve lineup) on Train A High Pressure Safety Injection System valves.

A second operator has just been dispatched to perform a Valve Position Verification on the system.

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 32 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) 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. Check the locking device secured for 3SIH*V107 and that the locking device hinders valve operation.
b) Verify the sealing device is intact for 3SIH*V89. Remove the locking device for 3SIH*V107, CLOSE the valve, OPEN the valve the required number of turns and reinstall the locking device.
c) Verify the sealing device is intact for 3SIH*V89. Check the locking device secured for 3SIH*V107 and that the locking device hinders valve operation.
d) 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. 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 sealed Back-seated Open. This is verifying the sealing device is intact and has NOT been damaged ("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): (Attach if not previously provided, including version/revision number)	PI-AA-500 (Rev. 3), Attachment 2, step 1.d and e (Sealed/Locked valves) OP 3308-003 (Rev. 005-03)

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, D) in-service breakers, E) disabled breakers, and F) removed from service breakers	(As available)

Question Source:	Bank #85268
Question History:	Millstone 3 2009 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	55.41.10
Comments:	

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

In accordance with AD-AA-102, *Procedure Use and Adherence*, what specific requirements exist while executing steps from a "Reference Use" procedure?

a) The operator will have a copy of the procedure or applicable pages available at the work site or be in direct communication with someone who has a copy and refer to the procedure as often as required to complete the task in accordance with the procedure requirements.
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 either have the applicable pages of the procedure in hand or be in direct communication with someone who has the procedure and placekeep as each step is performed.
d) The operator will review the procedure, as needed, before using it to perform the task. However, performing the activity from memory does NOT relieve the user of the responsibility of performing it in accordance with the procedure.

Proposed Answer: A

Explanation (Optional): "A" is correct, since a Reference Use procedure shall be used as follows:
1. HAVE a copy of the procedure or applicable pages available at the work site OR BE in direct communication with someone who has a copy.
2. REFER to the procedure as often as required to complete the task in accordance with the procedure requirements.
"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): AD-AA-102 (Rev. 12-0), Section 3.2.1.b
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06763 State the criteria that would assign a procedure as a Reference Level of Use (As available)

Question Source: Bank #85267

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 69	Tier #	3	3
K/A Statement: Ability to recognize system parameters	Group #	2	2
that are entry level conditions for Tech Specs	K/A #	GEN.2.2.42	
Proposed Question:	Importance Rating	3.9	4.6

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

- SIL Accumulator "D" borated water volume is 6650 gallons.
- RWST Recirculation Pumps (3QSS-P1A and P1B) are NOT running.
- RWST borated water volume is 1,100,000 gallons.
- RWST temperature is 46° F.

Which of these conditions requires entry into a Technical Specification ACTION STATEMENT?

a) Accumulator Volume.
b) RWST Recirc Pumps.
c) RWST Volume.
d) RWST Temperature.

Proposed Answer: C

Explanation (Optional): "A" is wrong, but plausible, since accumulator volume is within the required volume of 6618 and 7030 gallons. "B" is wrong, since RWST Recirc Pumps cycle to maintain RWST temperature, but are not required by Technical Specifications. "C" is correct, since RWST volume is below the required volume of 1,166,000 and 1,207,000 gallons. "D" is wrong, since RWST temperature is between within the required temperature band of 42 and 73°F. “A” and “D” are plausible, since these are Tech Spec requirements. “B” is plausible, since RWST Recirc Pumps maintain RWST Temperature, which is required by Tech Specs.

Technical Reference(s): Tech Spec LCO 3.5.1 (Amendment 258)
(Attach if not previously provided, Tech Spec LCO 3.5.4. (Amendment 262)
including version/revision number)

Proposed references to be provided to applicants during examination:

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

Question Source: Modified Bank #74365 Parent Question attached

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:
Question is considered "modified" since RWST Level and Accumulator Volume have been changed in the stem, and RWST Recirc Pump status has been added to the stem. There is a new correct answer, and a new distractor about RWST Recirc Pumps.

Original Bank #74365

With the plant at 100% power, the RO is taking rounds and observes the following parameters:

- SIL Accumulator "D" borated water volume is 6200 gallons.
- SIL Accumulator "D" pressure is 640 psia.
- RWST borated water volume is 1,200,000 gallons.
- RWST temperature is 45° F.

Which of these conditions requires entry into a Technical Specification ACTION STATEMENT?

e) Accumulator volume.
f) Accumulator pressure.
g) RWST volume.
h) RWST temperature.

Correct Answer: A

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 70	Tier #	3	3
K/A Statement: Ability to interpret Control Room indications to	Group #	2	2
verify the status and operation of a system, and understand how	K/A #	GEN.2.2.44	
operator actions and directives affect plant and system conditions	Importance Rating	4.2	4.4

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

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

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

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

a) The FW PUMPS P4 TRIP BYPASS Switch should be in BYPASS.
b) The Steam Dump "MODE SEL" Switch should be in TAVE Mode.
c) The Atm Relief Bypass Valve Lockout Switch should be in NORMAL.
d) The Feed Isolation Valve position indicator should indicate RED.

Proposed Answer: A

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

Technical Reference(s): OP 3203 (Rev. 23-0), steps 4.3.57, 58, and 64
(Attach if not previously provided, OP 3204 (Rev. 30-0), step 4.1.15
including version/revision number) GA-26 (Rev. 03-0), step 8

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #86763

Question History: Millstone 3 2013 NRC Exam

Question Cognitive Level: Comprehension or Analysis

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

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 71	Tier #	3	3
K/A Statement: Ability to control	Group #	3	3
radiation releases	K/A #	GEN.2.3.11	
Proposed Question:	Importance Rating	3.8	4.3

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

What two subsequent actions are required to be performed to check for potential sources of the radioactive release?

a.) Check the Condenser Air Ejector Discharge Radiation Monitor 3ARC-RE21 trend, and check the CTMT Vacuum Pumps running.
b.) Check the Condenser Air Ejector Discharge Radiation Monitor 3ARC-RE21 trend, and check the CTMT Purge Exhaust Fans running.
c.) Check the RPCCW Radiation Monitor 3CCP-RE31 trend, and check the CTMT Purge Exhaust Fans running.
d.) Check the RPCCW Radiation Monitor 3CCP-RE31 trend, and check the CTMT Vacuum Pumps running.

Proposed Answer: A

Explanation (Optional): "A" is correct since both the Condenser Air Ejectors and the CTMT Vacuum Pumps discharge to Gaseous Waste, which goes to the Millstone Stack. "B" and "C" are wrong, since the Containment Purge System exhausts to the Turbine Bldg stack, but plausible since it draws on CTMT, as does the CTMT vacuum pumps. "C" and "D" are wrong, since RPCCW is monitored by CCP-RE31, which is not alarming. "C" and "D" are plausible, since RPCCW overflows to the Aux Bldg, which is drawn on by SLCRS.

Technical Reference(s): AOP 3573 (Rev 23-0), Attachment A, Page 10 of 12
(Attach if not previously provided, including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05929 Describe the major action categories within AOP 3573. (As available)

Question Source: Bank #78805

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10 and 41.11

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 72	Tier #	3	3
K/A Statement: Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions	Group #	3	3
	K/A #	GEN.2.3.14	
	Importance Rating	3.4	3.8

Proposed Question:

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

1. Area Radiation Monitor 3RMS16-1 (VCT and Boric Acid Tank area) goes into alarm.
2. The RO reviews the narrative log and notes the following evolutions have recently been conducted:

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

Which of these activities caused the alarm?

a) The shutdown of the Degasifier.
b) The Liquid Waste discharge.
c) The startup of the Boron Evaporator.
d) The resin transfer.

Proposed Answer: A

Explanation (Optional): The degasifier degasses the letdown stream prior to entry into the VCT. With the degasifier shutdown, radioactive gasses accumulate in the VCT on the Aux Building 43 foot level near Area Rad Monitor 3RMS-RE16-1, so a RMS16-1 alarm can be anticipated ("A" correct). "B" is wrong, since the liquid waste discharge piping runs through the Aux Building 4 foot level. "C" is wrong, since the Boron Evaporator receives water from the Boron Recovery Tanks, which contains water that has settled and degassed over time prior to being sent to the degasifier. "D" is wrong, since the resin transfer path from the solid waste system runs through the Auxiliary Building, 4 foot level. "A", "C", and "D" are plausible, since they involve movement of radioactive liquid through the Auxiliary Building.

Technical Reference(s): AOP 3573 (Rev 23-0), Att B, page 2 of 5
(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, including the basis for each. (As available)

Question Source: Bank #76167

Question History: 2015 NRC Exam

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
K/A Statement: Knowledge of EOP entry	Group #	4	4
conditions and immediate action steps	K/A #	GEN.2.4.1	
Proposed Question:	Importance Rating	4.6	4.8

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

1. The D RCP HI HI VIBRATION Annunciator is received on MB4.
2. The US directs the RO to trip the reactor, stop the "D" RCP, and enter E-0, *Reactor Trip or Safety Injection*.
3. The RO attempts to trip the reactor from MB4, but the reactor does NOT trip.
4. The BOP operator attempts to trip the reactor from MB7, but the reactor does NOT trip.
5. The BOP operator attempts to trip the Load Center Supply breakers from Load Centers 32B and 32N, but the breakers supplying Load Center 32N do NOT open.
6. The RO verifies Rods are inserting in Automatic.
7. The BOP operator attempts to trip the Main Turbine, but the Turbine does NOT trip.

What is/are the next required action(s) to be taken by the RO and/or BOP operators?

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

Proposed Answer: B

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

Technical Reference(s): OP 3353.MB4B (Rev 08-0), 3-7, step 3
(Attach if not previously provided, FR-S.1 (Rev. 20-0), Caution prior to step 1, and steps 1 and 2
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04625 Describe the major action categories within EOP 35 FR-S.1. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 74	Tier #	3	3
K/A Statement: Knowledge of operator	Group #	4	4
response to loss of all annunciators	K/A #	GEN.2.4.32	
Proposed Question:	Importance Rating	3.6	4.0

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

1. The ANN PWR SPLY MB1, 2, 3, 4, 5 FAILURE annunciator is received on Main Board (MB) 6.
2. The RO reports annunciators are lost to MB 1, 2, 3, and 4.
3. The BOP reports annunciators are lost on MB 5, but annunciators are still functioning on MB 6, 7, 8, and VP1.
4. The US enters AOP 3574, *Loss of Main Board Annunciation.*

Complete the following statements concerning two actions that the crew will be directed to take per AOP 3574.

Dispatch a PEO to check Annunciator power from __(1)__. Establish augmented visual monitoring of the affected Main Boards by assigning __(2)__.

a) (1) Battery 5 (3BYS-PNL 5-1 and 5-2)
   (2) one extra licensed operator designated to monitor the five affected MB sections
b) (1) Battery 5 (3BYS-PNL 5-1 and 5-2)
   (2) more than one licensed operator, each designated to monitor a maximum of two MB sections
c) (1) Battery 6 (3BYS-PNL 6-1 and 6-2)
   (2) one extra licensed operator designated to monitor the five affected MB sections
d) (1) Battery 6 (3BYS-PNL 6-1 and 6-2)
   (2) more than one licensed operator, each designated to monitor a maximum of two MB sections

Proposed Answer: B

Explanation (Optional): AOP 3574 will dispatch a PEO to check annunciator power supplies at 3BYS-PNL 5-1 and 5-2. "C" and "D" are wrong, but plausible, since Battery 6 is the other non-vital Battery, located in the Turbine Building. "B" is correct, and "A" wrong, but plausible, since AOP 3574 will establish augmented Main Board monitoring, requiring licensed operators to monitor a maximum of two Main Board sections each.

Technical Reference(s): AOP 3574 (Rev. 05-0), steps 2 and 4, and Attachment A
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05934 Describe the major action categories within AOP 3574 (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 75	Tier #	3	3
K/A Statement: Ability to recognize abnormal	Group #	4	4
indications for system operating parameters which	K/A #	GEN.2.4.4	
are entry-level conditions for EOPs and AOPs	Importance Rating	4.5	4.7

Proposed Question:

Which one of the following conditions would require the crew to enter AOP 3566, *Immediate Boration*?

a) The ROD CONTROL BANKS LIMIT LO annunciator is received during a rapid downpower.
b) At 100% power, an unexplained event is causing reactor power and Tave to increase.
c) On a Safety Injection from 100% power, two Control Rods do NOT insert.
d) The RCS cools down uncontrollably to 540°F on a reactor trip.

Proposed Answer: B

Explanation (Optional): "B" is correct, since this is an indication of an unexplained positive reactivity addition. "A" is wrong, but plausible, since Rod LO-LO is the Immediate Borate setpoint, and AOP 3575 addresses a Rod Lo-Lo condition without requiring entry into AOP3566. "C" is wrong, since immediate boration is not required on a safety injection. "C" is plausible, since immediate boration is required for 2 or more stuck rods on a reactor trip without safety injection. "D" is wrong, but plausible, since the setpoint requiring entry into AOP 3566 is 530°F.

Technical Reference(s): AOP 3566 (Rev. 13-0), Entry Conditions
(Attach if not previously provided, ES-0.1 (Rev. 29-0), step 6
including version/revision number) AOP 3575 (Rev. 26-0), step 7

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #64275

Question History: Millstone 3 2000 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 76	Tier #		1
K/A Statement: Ability to verify alarms are consistent	Group #		1
with plant conditions during a Pzr vapor space accident	K/A #	APE.008.GEN.2.4.46	
Proposed Question:	Importance Rating		4.2

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

1. A reactor trip and safety injection occurs and the crew is in E-1, *Loss of Reactor or Secondary Coolant.*
2. RCS pressure is 2350 psia, with pressure being controlled by the PZR PORV's.
3. SG pressures are 1100 psig and stable.
4. The crew is preparing to perform E-1, step 9 "Check RCS and SG Pressures" when the RO reports RCS pressure is 2200 psia and lowering.
5. The RO reports "A" PORV indicates "dual indication".
6. The BOP reports EEQ tailpiece temperature for the "A" PORV has stabilized at 176°F.

Complete the following statement for the cause of stable PORV tail piece temperatures and the procedural action required to be taken.

The tailpiece temperatures have stabilized due to ___(1)___. The crew is required to___(2)___ .

a) (1) the design of the EEQ temperature monitor
   (2) close the "A" PORV, and if the "A" PORV will not close, close its block valve
b) (1) the design of the EEQ temperature monitor
   (2) return to E-1, step 1, due to stable SG pressures (without closing the "A" PORV)
c) (1) the thermodynamic behavior of the leaky PORV
   (2) close the "A" PORV, and if the "A" PORV will not close, close its block valve
d) (1) the thermodynamic behavior of the leaky PORV
   (2) return to E-1, step 1, due to stable SG pressures (without closing the "A" PORV)

Proposed Answer: A

Explanation (Optional): The EEQ tailpiece temperature has a maximum range of 176°F. There is a NOTE in the associated ARP, MB4A 3-5, Pressurizer Relief Valve Dis Temp Hi, which identifies this condition. "C" is wrong, since if the EEQ limitation didn't exist, the constant enthalpy throttling process would yield a temperature of ~220°F (for 18 psia in the PRT). "C" is plausible, since the constant enthalpy process results in a tailpiece temperature that is significantly lower than Pressurizer temperature (Lesson learned for Three Mile Island event). "A" is correct, since per the caution prior to step 5, step 5 is a continuous action step. Step 5a and its RNO will close the PORV, and if the PORV fails to close, close its block valve. "B" and "D" are wrong, since if step 9 were commenced without closing the PORV, it would direct the operators to step 10, which will end up transitioning to ES-1.2 at step 12, which is not desired. By closing the PORV, the crew will loop back to step 1, and transition to ES-1.1 at step 6. "B" and "D" are plausible, since if step 5 were not considered a continuous action step, the crew would commence step 9, which contains a loop back to step 1, which would address the PORV when step 5 is reached.

Technical Reference(s):	E-1 (Rev. 26-0), step 5 (including caution), and step 9
(Attach if not previously provided,	OP 3353.MB4A (Rev 05-0), 3-5
including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-04914 Outline the unique characteristics of a Pressurizer Vapor Space LOCA	(As available)

Question Source:	New
Question History:	
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.5, 41.7, and 43.5
Comments:	

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 77	Tier #		1
K/A Statement: Determine/interpret location	Group #		1
and isolability of leaks during a loss of RHR	K/A #	APE.025.AA2.4	
Proposed Question:	Importance Rating		3.6

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

- The "A" Train of RHR is in operation in the "Single RHR Loop" alignment.
- The "A" Train is the protected "Operable" train.

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

1. Offsite power is lost.
2. Pressurizer level starts to drop.
3. Neither EDG starts.
4. The crew enters EOP 3501, *Loss of All AC Power (MODE 5, 6, and Zero).*
5. Per EOP 3501, step 1, the US directs the RO to isolate the RCS to minimize inventory loss.
6. The crew successfully starts the "A" EDG from MB8 per EOP 3501, step 3, and restores power to bus 34C.
7. The crew verifies the "A" CHS Pump is running.
8. A PEO reports a significant leak is coming from the CHS piping at the entrance to the Letdown Heat Exchanger in the Letdown Heat Exchanger Room.

Complete the following statement concerning whether or not the RO's actions taken to isolate the RCS successfully isolated RCS from the leak, and which procedure the US will use to restore shutdown cooling.

The RO's actions in EOP 3501, step 1 __(1)__ successfully isolate the RCS from the leak, and the US will remain in/transition to __(2)__ to restore shutdown cooling.

	(1)	(2)
a)	DID	EOP 3501, *Loss of All AC Power (MODE 5, 6, and Zero)*
b)	DID NOT	EOP 3501, *Loss of All AC Power (MODE 5, 6, and Zero)*
c)	DID	EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory)*
d)	DID NOT	EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory)*

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires detailed knowledge of the RCS isolation step in EOP 3501, and also requires the applicant to determine whether transition from EOP 3501 to 3505 is required under current conditions, and both procedures will attempt to restore shutdown cooling. When power is restored to the Operable AC emergency bus from its emergency diesel or offsite power, EOP 3501 directs the crew to EOP 3505 ("A" and "B" wrong). "A" and "B" are plausible, since the crew will directed to establish shutdown cooling in EOP 3501 if power is not restored, or is restored to a degraded bus, or is restored by the SBO Diesel. EOP 3505 will attempt to isolate the leak by closing 3CHS-HCV128 and all three letdown orifice isolation valves. "C" is correct, and "D" wrong, since the Letdown Heat Exchanger is downstream of the Letdown Isolation Valves, and downstream of HCV128. This action will successfully isolate the leak from the RCS. "D" is plausible, since there are significant portions of piping that are not isolated by these actions.

Technical Reference(s):	EOP 3501 (Rev. 20-0), step1 and step 2.a, b, and l
(Attach if not previously provided,	EOP 3505 (Rev. 15-0), entry condition 2.a, and step 5.RNO.e
	P&ID 104A (Rev. 54)
including version/revision number)	P&ID 112A (Rev. 50)

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 History:

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 # 78	Tier #		1
K/A Statement: Ability to verify system alarm	Group #		1
setpoints and operate controls identified in the Alarm	K/A #	APE.056.GEN.2.4.50	
Response Manual during a loss of offsite power	Importance Rating		4.0

Proposed Question:

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

1. Offsite power is lost.
2. The crew completes step 4 of E-0, *Reactor Trip or Safety Injection*.
3. As part of the transition brief, the US directs the RO and BOP to report the status of Main Board annunciators.
4. The BOP operator reports the following two annunciators are lit:
   - TURB BRG L/O HEADER PRES LO (MB7A, 1-8).
   - GEN EMER SEAL OIL PP RUNNING (MB7A, 4-5).
5. The BOP Operator checks the expected Lube Oil/Seal Oil Pumps running, and reports the following:
   - Bearing header pressure is decreasing.
   - The Emergency Seal Oil Pump is running.

Which of these two annunciators is UNEXPECTED, and what action will the US direct per the appropriate Annunciator Response Procedure?

a) GEN EMER SEAL OIL PP RUNNING. The US will direct the BOP to momentarily stop the Emergency Seal Oil Pump to attempt to reseat the Seal Oil Relief Valve.
b) GEN EMER SEAL OIL PP RUNNING. The US will direct a PEO to vent Main Generator Hydrogen to atmosphere to lower Generator hydrogen pressure.
c) TURB BRG L/O HEADER PRES LO. The US will direct a PEO to use GA-1, *Energizing MCC 32-3T*, to re-energize MCC 32-3T and start the Lift Oil Pumps.
d) TURB BRG L/O HEADER PRES LO. The US will direct the BOP to close the MSIVs and break Condenser vacuum.

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions and select entry into the appropriate Annunciator Response Procedure (ARP), followed by selecting a specific procedure section to mitigate the event and determining the action that will be required per the ARP, which goes beyond system knowledge. Also, each of the specified actions are directed in ARPs or EOPs and are related to seal oil and lube oil. On a loss of offsite power, power is lost to the Main Seal Oil Pump, which is AC powered, and to the AC powered Lube Oil Pumps. Seal Oil and Lube Oil have DC Powered emergency pumps, and Lube Oil has a Main Shaft Oil Pump driven by the Main Turbine Shaft, and an oil-driven booster pump. The DC Emergency Seal Oil Pump automatically starts if the Main Seal Oil Pump stops ("A" and "B" wrong). "A" and "B" are plausible, since normally on a reactor trip without a LOP, neither of these annunciators are expected, and the ARP does cycle the Emergency Seal Oil Pump to attempt to reseat the check valve on low pressure. Also, the Seal Oil to Hydrogen DP Lo ARP directs a PEO to vent the Main Generator Hydrogen to atmosphere if Seal Oil pressure is low. Lube oil pressure decreasing is plausible, since on a turbine trip, the Main Shaft Oil Pump, which is driven by the Main Turbine, will start slowing down, resulting in lowering Lube Oil Pressure, but backup pumps will start before the low pressure alarm is received. "D" is correct, since the Lube Oil Low Pressure ARP directs the crew to close MSIVs and break vacuum. "C" is wrong, since the ARP does not direct starting the Lift Oil Pumps. "C" is plausible, since Lift Oil Pumps normally start to increase bearing oil pressure after a turbine trip when the Turning Gear Oil Pump starts. These pumps are directed to be started following a turbine trip in other procedures using GA-1.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	OP 3353.MB7A (Rev. 07-0), 1-8 and 4-5
	P&ID 141A (Rev. 22)
	P&ID 141B (Rev. 24)
	P&ID 141C (Rev. 12)
	P&ID 142A (Rev. 32)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05754 ...describe the turbine lube oil system flow path and alignment under the following normal, abnormal, and emergency conditions... Loss of AC power... (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55. 41.4 and 43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 79	Tier #		1
K/A Statement: Determine/interpret the existence of a SG tube rupture and its potential consequences	Group #		1
	K/A #	EPE.038.EA2.02	
Proposed Question:	Importance Rating		4.8

Safety Injection actuates on Low Pressurizer Pressure, and the following sequence of events occurs:

Time (Minutes)	Action
0	The crew enters E-0, *Reactor Trip or Safety Injection.*
12	The BOP reports the "A" SG level is increasing in an uncontrolled manner.
25	The BOP operator isolates AFW flow to the "A" SG with NR level at 31%.
27	The "A" SG Atmospheric Relief Valve fails open.
40	The BOP notices the failed open Atmospheric Relief Valve, and closes the valve.
45	Safety Injection is terminated.

Complete the following statement concerning "operator credited actions" assumed by the FSAR for this event.

The operators were too slow to___(1)___, and the potential adverse consequence due to this delay is___(2)___.

a) (1) isolate AFW Flow to the "A" Steam Generator
   (2) exceeding the 10CFR fuel clad temperature limit
b) (1) isolate AFW Flow to the "A" Steam Generator
   (2) rad release to the environment exceeding off-site dose estimates
c) (1) close the "A" SG Atmospheric Relief Valve
   (2) exceeding RCS boundary thermal stress limits
d) (1) close the "A" SG Atmospheric Relief Valve
   (2) rad release to the environment exceeding off-site dose estimates

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it requires knowledge of the FSAR accident analysis beyond system knowledge. The Millstone 3 license (10CFR55.43.1) requires compliance with Tech Specs, which are included in 10CFR55.43.2, and states that Millstone... Unit No. 3... is described in the licensees' 'Final Safety Analysis Report."... The FSAR assumes operators comply with certain action times to ensure the accident analysis is within acceptable limits, and the US is responsible for driving through the EOPs at a rate that will comply with these times. Based on low Pzr pressure and SG level increasing in an uncontrolled manner, this event can be diagnosed as a SGTR. Two operator credited actions that are required to be met to comply with the FSAR for the SGTR event are: a failed open atmospheric relief valve needs to be closed by the operators within 20 minutes of failing open (to remain within rad release assumptions), and isolating AFW flow to the ruptured SG by 30% Narrow range level, to comply with SG overfill analysis. Operators met the time requirement for the atmospheric relief valve ("C" and "D" wrong, but plausible), but did not meet the SG Level AFW requirement. The FSAR indicates that a significant partitioning factor that exists between SG water and SG steam is used in calculating off-site dose during the design basis SGTR. "B" is correct, and "A" wrong, since the basis for isolating AFW is to prevent SG overfill, and ECCS flow and RWST inventory will keep the core covered during a SGTR. "A" is plausible, since a SGTR is in progress, RCS inventory is decreasing, and 2200°F clad temperature is a 10CFR design criterion. Also, PTS is a concern in E-3 during the rapid cooldown if ruptured SG pressure is too low, but in this case, a transition to ECA-3.1would be required.

Technical Reference(s):	FSAR Chapter 15.6.3 (Rev. 30), Section 15.6.3.1
(Attach if not previously provided,	FSAR Chapter 15.6.3 (Rev. 30), Section 15.6.3.2.1
including version/revision number)	FSAR Chapter 15.6.3.2.2 (Rev. 30), Radiological Consequences
	FSAR Table 15.6.3-1 (Rev. 30) Operator Action Times

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04951 Outline the anticipated Operator Actions in response to SGTRs to include the operator credited actions in FSAR chapter 15. (As available)

Question Source: New

Question History:

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 # 80	Tier #		1
K/A Statement: Knowledge of operational	Group #		1
implications of EOP warnings, cautions, and	K/A #	WE04.GEN.2.4.20	
notes during a LOCA Outside Containment	Importance Rating		4.3

Proposed Question:

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 and cycles at 2350 psia, with PZR PORVs cycling.
3. The crew is responding using ECA-1.2, *LOCA Outside Containment.*
4. RWST level is 900,000 gallons and slowly decreasing.
5. Pressurizer level is 65% and increasing.
6. While attempting to isolate the break, the final valve the crew closes in attempt to isolate the break is RHR pump "A" cold leg injection valve (3SIL*MV8809A).
7. After 3SIL*MV8809A closes, the RO reports that RCS pressure is still cycling at 2350 psia.
8. The STA reports that the PORVs are cycling at the same rate as before the valve was closed.

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

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

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific Emergency Contingency procedures. "D" is correct and "A" wrong, since it can be determined that the break is not isolated with no change in the PORV cycling rate. "A" is plausible since step 5 directs transition to E-1 if pressure is increasing, and pressure can't increase due to the PORVs cycling. The NOTE prior to step 1 must be applied while pressure is cycling on the PORVs. "B" is plausible, since with the break isolated, the crew would transition to E-1 first, and then to ES-1.1. "C" is wrong because during a LOCA outside CTMT, a loss of recirculation capability exists. "C" is plausible, since this would normally be the correct transition on a small break LOCA, which is in progress.

Technical Reference(s): ECA-1.2 (Rev. 08-0), Note prior to step 1
(Attach if not previously provided, ECA-1.2 (Rev. 08-0), steps 4 and 5
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03870 Identify plant conditions that require entry into EOP 35 ECA-1.1. (As available)

Question Source: Bank #78931

Question History: Millstone 3 2004 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 81	Tier #		1
K/A Statement: Determine/interpret facility conditions and selection of appropriate procedures during a loss of secondary heat sink	Group #		1
	K/A #	WE05.EA2.1	
Proposed Question:	Importance Rating		4.4

The crew has entered E-0, *Reactor Trip or Safety Injection*, and the RO and BOP operators report the following:

- RCS Pressure: 1000 psia
- SG pressures: 1092 psig
- SG NR Levels: Offscale Low
- Ctmt Temperature: 145°F
- No AFW Pumps are running, and none can be started at MB5.

What procedure transition(s) will be required to address the loss of AFW Pumps?

a) At E-0, step 9, transition to FR-H.1, *Loss of Secondary Heat Sink*, where no actions to restore heat sink will be required. Transition from FR-H.1 back to E-0, step 10.
b) At E-0, step 9, transition to FR-H.1, *Loss of Secondary Heat Sink*. Then take actions per FR-H.1 to restore heat sink.
c) When CSF Status Tree monitoring is initiated at E-0, step 16, transition to FR-H.1, *Loss of Secondary Heat Sink*, where no actions to restore heat sink will be required. Transition from FR-H.1 to E-1, step 1.
d) When CSF Status Tree monitoring is initiated at E-0, step 16, transition to FR-H.1, *Loss of Secondary Heat Sink*. Then take actions per FR-H.1 to restore heat sink.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it requires assessing plant conditions and transitioning to appropriate procedure sections. E-0 will check for heat sink at step 9, and transition to FR-H.1 ("C" and "D" wrong). "C" and "D" are plausible, since the normal transition steps in E-0 occur in steps 14-16, when status tree monitoring is initiated. FR-H.1 will direct a transition back to procedure and step in effect, since RCS pressure is less than steam generator pressure. This indicates that adequate heat removal is occurring out the break, so a secondary heat sink is not required ("A" correct, "B" wrong). "B" is plausible, since this would be correct if RCS pressure were greater than SG pressure.

Technical Reference(s): E-0 (Rev. 32-0), steps 9 and 16
(Attach if not previously provided, FR-H.1 (Rev. 26-0), step 1
including version/revision number)

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #63990

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 82	Tier #		1
K/A Statement: Determine/interpret uncontrolled rod withdrawal from available indications	Group #		2
	K/A #	APE.001.AA2.5	
Proposed Question:	Importance Rating		4.6

Initial conditions:

- The plant is at 60% power, BOL.
- Control Bank "D" rods are at 160 steps.

The following sequence of events occurs:

1. With all other conditions stable, the RO reports Control Bank D rods are stepping out.
2. The RO places rods in MANUAL.
3. The "T REF/AUCT TAVE DEVIATION" annunciator comes in on MB4.

What initial action is the crew required to take to minimize the Tave/Tref deviation, and per the FSAR, what ANS Condition Event is in progress?

a) Adjust turbine load. ANS Condition II.
b) Adjust turbine load. ANS Condition III.
c) Adjust boron concentration. ANS Condition II.
d) Adjust boron concentration. ANS Condition III.

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions, select entry into the appropriate AOP, and determine the specific procedure action required to mitigate the event. Also, it requires knowledge of FSAR accident analysis beyond system knowledge. The Millstone 3 license (10CFR55.43.1) requires compliance with Tech Specs, which are included in 10CFR55.43.2, and states that Millstone… Unit No. 3... is described in the licensees' 'Final Safety Analysis Report'."… . This knowledge of FSAR ANS Accident Conditions assists the US in determining the likelihood of sustaining fuel damage during the event. An uncontrolled Control Bank D rod withdrawal event can be diagnosed, since Bank D Group position indication shows rods withdrawing, and this is confirmed by the Tref/Tave Deviation annunciator. When the rods withdraw, positive reactivity was being added to the core. This initially causes Reactor Power to increase, causing heat in to the RCS to exceed heat out via the SGs. This causes Tave to increase, which adds negative reactivity. An uncontrolled rod withdrawal event is an entry condition for 3552, *Malfunction of the Rod Drive System*, which directs the RO to place rod control in Manual. At this point, reactor power is near its original value, with an elevated Tave. Since it can be determined that Tave is now greater than Tref, AOP 3552 directs the crew to borate to minimize Tave-Tref deviation ("A" and "B" wrong). "A" and "B" are plausible, since raising Turbine Load would also lower Tave, and AOP 3552 would raise turbine load if the steam dumps had opened. It can be determined that the steam dumps did not open, since a load reject did not occur. AOP 3552 also would direct adjusting turbine load if Tave was less than Tref. "C" is correct, and "D" wrong, since an uncontrolled Bank withdrawal is classified as an ANS Condition II Event, since RPS will terminate any bank withdrawal event before DNBR falls below the safety analysis limit value. "D" is plausible, since a single rod withdrawal event is classified as a Condition III event, since an automatic reactor trip may not occur fast enough to prevent the minimum DNBR limit from being violated in localized areas in the core. Also, the FSAR does not assume the RO places rods in manual, and rods started fairly low in the core, meaning the FSAR event would be significantly worse, with the entire bank of rods continuing to withdraw until automatic reactor trip occurs. Thus the FSAR event is much worse than the actual event with prompt operator action.

Technical Reference(s): AOP 3552 (Rev. 14-0), Entry Condition 2.1
(Attach if not previously provided, AOP 3552 (Rev. 14-0), step 1.a through 1.e
including version/revision number) FSAR, Chapter 15.4.2.1 (Rev. 30)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04898 Identify the specific types of events analyzed as Reactivity & Power Distribution Anomalies (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1, 41.5, 43.1, and 43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 83	Tier #		1
K/A Statement: Knowledge of the emergency action level	Group #		2
thresholds and classifications during high RCS activity	K/A #	APE.076.GEN.2.4.41	
Proposed Question:	Importance Rating		4.6

With the plant initially at 100% power with Containment High-Range Monitor 3RMS-RE04A out of service, the following sequence of events occurs:

Time	Event
0 Minutes:	A locked rotor occurs on the "A" RCP, resulting in a reactor trip.
5 Minutes:	While in ES-0.1, *Reactor Trip Response*, the STA reports Fuel Drop Monitors 3RMS-RE 41 and 42 are reading 0.5 R/hr and trending up.
10 Minutes:	A Yellow Path is received on CTMT Radiation.
120 Minutes:	3RMS- RE 41/42 are reading 150 R/hr and continuing to trend up.
120 Minutes:	Containment High-Range Monitor 3RMS- RE 05A is reading 150 R/hr and trending up.
145 Minutes:	3RMS- RE 05A and 3RMS- RE 41/42 all read 175 R/hr and stable.
149 Minutes:	RMT1 reports the Core Damage Estimate is NOT yet complete.
150 Minutes:	A fault occurs on the "C" Steam Generator inside Containment, and the "C" SG rapidly depressurizes.
151 Minutes:	3RMS-RE-05A rapidly increases to 1000 R/hr.
151 Minutes:	3RMS-RE 41 and 42 readings have remained stable at 175 R/hr.

Using the Millstone 3 EAL Tables, what is the highest classification required for this event?

a) General Emergency – Alpha
b) Site Area Emergency – Charlie 2
c) Alert – Charlie 1
d) Unusual Event – Delta 1

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions related to radiation readings and make an emergency classification, which is a duty reserved for a SRO licensed individual. The Alert classification is first met at 10 minutes for "In-plant Radiation" (RA2), Radiation reading >5R/hr in Areas Requiring Access for Safe Shutdown (CTMT), since the Yellow path for CTMT Radiation is 10 R/hr, and for Loss of the Clad Barrier (FCB3) with sustained valid RE-04A/05A reading >5R/hr without RCS release inside CTMT. At 121 minutes, the classification remains Alert, C-1, even though CTMT radiation exceeds the Table 1 "2 to 4" hour limit of 125 R/hr, since this only applies with an RCS leak inside CTMT. "C" is correct, and "A", "B", and D" wrong, since at 151 minutes, the classification does not escalate, since the CTMT loss threshold is still not exceeded. "A" is plausible, since RE05A has rapidly increased above the RG1 "In-Plant Radiation" General Emergency threshold, but this is not valid, since the steam break inside CTMT is causing Temperature-Induced Current (TIC) on RE05A, and its readings are not to be considered valid until the effects of TIC have dissipated. Also, radiation impacts all three barriers are the barrier failure reference table. "B" is plausible, since this would be correct if the applicant uses the TIC-affected reading of RE05A to assess the CNB5 CTMT loss threshold, making two barriers lost, or, if the applicant views the RCS barrier as potentially lost due to uncontrolled pressure decrease (due to the steam break) and increasing containment radiation monitors (RCB3), which are elevated, but the rad monitor indication was stable during the steam break. "D" is plausible, since a steam break inside CTMT is not classifiable if Radiation levels were lower, if only the CTMT barrier is lost, Unusual Event would be correct, and unexpected RMS reading increasing by >1000 times normal readings is an Unusual Event (RU1).

Technical Reference(s): MP-26-EPI-FAP06-003 (Rev. 11-0) EAL Tables
(Attach if not previously provided,
including version/revision number)

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

Learning Objective: EP-00171 Given a plant condition and associated alarms and/or indications, classify an emergency event to include NRC classification and state posture code (As available)

Question Source: Bank #89294

Question History: Millstone 3 2013 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 # 84	Tier #		1
K/A Statement: Determine/interpret facility conditions and selection of appropriate procedures during LOCA cooldown and depressurization	Group #		2
	K/A #	WE03.EA2.1	
	Importance Rating		4.2

Proposed Question:

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

1. At the transition brief from E-0, *Reactor Trip or Safety Injection*, the RO reports the following:
   - The Pressurizer is empty.
   - CDA has NOT actuated.
2. The crew transitions to E-1, *Loss of Reactor or Secondary Coolant* due to abnormal radiation on 3CMS*RE22 (pre-trip).
3. The crew trips RCPs at E-1, Step 1, due to the RO's report that RCS pressure has slowly decreased to 1480 psia.

Assuming plant conditions do NOT significantly change, to what procedure will the crew be required to transition to from E-1?

a) ES-1.1, *SI Termination.*
b) ES-1.2, *Post LOCA Cooldown and Depressurization.*
c) ES-1.3, *Transfer to Cold Leg Recirculation.*
d) ES-1.4, *Transfer to Hot Leg Recirculation.*

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to event-specific Emergency Contingency procedures. "A" is wrong, but plausible, since this would be correct if PZR level was above 16%, and RCS pressure stabilizes. "B" is correct, and "C" and "D" wrong, since RCS pressure is above RHR shutoff head, and without CTMT spray running, RWST level will remain above 520,000 gallons for a long time. "C" and "D" are plausible, since entry to ES-1.3 and then 1.4 would be required if RHR was injecting or if RWST Lo-Lo was reached.

Technical Reference(s): E-1 (Rev. 26-0), Steps 1, 6, 12, 14, and 23
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-04363 Discuss conditions which require transition to other procedures from EOP 35 E-1 (As available)

Question Source: Bank #63935

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 85	Tier #		1
K/A Statement: Knowledge of the parameters and	Group #		2
logic used to assess the status of safety functions	K/A #	WE14.GEN.2.4.21	
during a loss of containment integrity	Importance Rating		4.6

Proposed Question:

A LOCA has occurred, and Initial Conditions are as follows:

1. The crew has completed E-0, *Reactor Trip or Safety Injection.*
2. The crew is performing a manual status tree check prior to transitioning to E-1, *Loss of Reactor or Secondary Coolant.*
3. The crew has verified the first four status trees are either Green or Yellow.
4. CTMT parameters currently indicate as follows:

- Containment pressure: 25 psia and slowly increasing.
- Containment radiation: 12 R/hr and slowly increasing.
- Containment sump level: 13 feet and slowly increasing.
- Quench Spray Pumps: Both running.

Complete the following statement.

The color of the Containment CSF Status Tree is ___(1)___, and the US will transition to ___(2)___.

	(1)	(2)
a)	Orange	FR-Z.1, *Response to High Containment Pressure*
b)	Orange	FR-Z.2, *Response to Containment Flooding*
c)	Orange	FR-Z.3, *Response to High Containment Radiation Level*
d)	Yellow	E-1, *Loss of Reactor or Secondary Coolant*

Proposed Answer: D

Explanation (Optional): This question is considered SRO level since it requires knowledge of diagnostic steps and decision points in the EOPs that involve transitions to Functional Restoration procedures, including yellow paths. It also involves EOP rules of use based on status tree colors. This question is considered a KA match, since the Containment Status Tree is designed to protect the Containment Fission Product Barrier (Containment Integrity concern), and Containment pressure, Sump level, and Radiation level all go into this effort. "D" is correct, since Ctmt orange paths are from Ctmt pressure of 23 psia with no Ctmt Spray Pumps running. "A" is wrong, since with Ctmt pressure above 23 psia with the Ctmt spray pumps running, the tree color based on Ctmt pressure is yellow. Per EOP rules of use, yellow paths are optional, and the EOP series procedures take priority over yellow path procedures. With a LOCA in progress, entry into E-1 is desired with Ctmt spray operating as designed. "A" is plausible, since Ctmt pressure is above 23 psia. "B" is wrong, since Ctmt sump level is below the high sump level setpoint of 15.75 feet. "B" is plausible, since Ctmt High Sump Level is an Orange path procedure entry condition, and sump level is abnormally high. "C" is wrong, since High Ctmt Radiation is a yellow path procedure. "C" is plausible, since Ctmt radiation exceeds the 10R/hr high radiation entry condition.

Technical Reference(s):	EOP 35 F-0.5 (Rev. 04-0), Ctmt CSF Status Tree
(Attach if not previously provided,	OP 3272 (Rev. 09-0), Attachment 4, pages 4 and 5 of 7
including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective: MC- 04666 Identify plant conditions which require entry into EOP 35 FR-Z.1 (As available)

Question Source: Bank #63966

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
K/A Statement: Predict impact/mitigate RHR System	Group #		1
pressure transient protection during cold shutdown	K/A #	005,A2.2	
Proposed Question:	Importance Rating		3.7

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

- The "A" PORV block valve is closed due to a leaky PORV.
- When RCS hot leg temperature reached 340°F, the "A" train of RHR was placed in service in the cooldown mode.
- When RCS hot leg temperature reached 230°F, the "B" train of RHR was also placed in service in the cooldown mode.
- Hot leg temperatures are 220°F.

The crew isolates the "B" train of RHR from the RCS due to a significant RHR piping leak.

Complete the following statement concerning the status of RCS overpressure protection.

Adequate RCS overpressure protection___(1)___available, since___(2)___.

	(1)	(2)
a)	is NOT	the crew was relying on both RHR suction reliefs to meet COPPS requirements
b)	is NOT	the crew was relying on the two PORVs to meet COPPS requirements
c)	IS	COPPS is NOT required at this RCS temperature, and one PORV is available
d)	IS	one PORV and one RHR Suction Relief are still available

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess detailed procedural requirements for arming COPPS, and Tech Spec LCO requirements in lower Modes, and compare them to abnormal plant conditions. COPPs is required <226°F ("C" wrong), from either 2 PORVs, 2 RHR suction relief valves, or one of each ("B" wrong). "D" is correct, and "A" wrong, since COPPS is armed by procedure when hot leg temperatures decrease to 250°F, and at this point, only one RHR Train was in service. "A" is plausible, since OP 3208 does NOT require the crew to ARM COPPS at 250°F if both trains of RHR are available in the cooldown mode during initial plant cooldown to MODE 5, since two RHR suction reliefs provides adequate COPPS protection. Also, the second train of RHR was placed in service prior to being needed per Tech Specs at 220°F. "B" is plausible, since normally, two RHR suction relief valves or two PORVs are available for COPPS. "C" is plausible, since COPPS is only required at cold temperatures.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	OP 3208 (Rev. 27-0), Note 3 prior to step 4.3.1, and step 4.3.5 OP 3208 (Rev. 27-0), Notes and Caution prior to step 4.3.34 Tech Spec LCO 3.4.9.3 (Amendment 197)

Proposed references to be provided to applicants during examination: None

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

Question Source:	Bank #85273
Question History:	Millstone 3 2009 NRC Exam
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.43.2 and 43.5
Comments:	

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 87	Tier #		2
K/A Statement: Ability to locate and operate comp-	Group #		1
onents, including local controls for the CCW System	K/A #	008.GEN.2.1.30	
Proposed Question:	Importance Rating		4.0

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

1. The "RPCCW HX SW FLOW HI/LO" annunciator is received on Main Board 1.
2. A PEO reports an unisolable Service Water piping rupture in the Aux Building, upstream of RPCCW Heat Exchanger Service Water Inlet Isolation Valve 3SWP*MOV50A.
3. The crew starts both Train "B" Service Water Pumps.

Complete the following statements concerning procedural direction to address the pipe break, and which procedure directs action by a PEO.

The US will enter__(1)__to place both Train "A" Service Water Pumps in Pull-To-Lock and maintain 3SWP*MOV50A open. Per__(2)__a PEO will be dispatched to check/realign RPCCW Process Radiation Monitor 3CCP-RE31 to the "B" Train.

a) (1) RPCCW HX SW FLOW HI/LO ARP
(2) AOP 3560, *Loss of Service Water*

b) (1) RPCCW HX SW FLOW HI/LO ARP
(2) AOP 3561, *Loss of Reactor Plant Component Cooling Water*

c) (1) AOP 3560, *Loss of Service Water*
(2) AOP 3560, *Loss of Service Water*

d) (1) AOP 3560, *Loss of Service Water*
(2) AOP 3561, *Loss of Reactor Plant Component Cooling Water*

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions and determine whether the AOP or ARP will be the appropriate entry point for this event, when both apply. This question also requires the applicant to understand AOP transition points for the specific event in progress. AOP 3560 is written assuming SWP pipe breaks have been addressed per the ARPs prior to entering the AOP 3560 will not mitigate the leak without having first taken ARP actions ("C" and "D" wrong). The RPCCW HX SW FLOW HI/LO ARP will direct the affected train Service Water Pumps to be placed in Pull-To-Lock and 3SWP*MOV50A will be left open since the break is upstream of this valve. After taking these steps per the ARP, the crew will enter AOP 3560 ("A", "C", and "D" plausible). AOP 3560 will transition to AOP 3561 prior to address the loss of cooling to RPCCW ("A" wrong). AOP 3561 will require a PEO to check/realign RPCCW Process Radiation Monitor 3CCP-RE31 to the "B" Train ("B" correct).

Technical Reference(s): OP 3353.MB1C (Rev. 12-0), 1-1A, step 8.3.1 through 8.3.5
AOP 3560 (Rev. 11-0), steps 1 and 2
AOP 3561 (Rev. 18-0), step B.4.g

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07541 Given a set of plant conditions, determine the required actions to be taken per AOP 3560. (As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4 and 43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 88	Tier #		2
K/A Statement: Predict impact/mitigate flowpaths	Group #		1
of Main Steam System steam during a LOCA	K/A #	039.A2.1	
Proposed Question:	Importance Rating		3.2

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

1. SIS actuates due to low Pressurizer pressure.
2. The BOP Operator reports one "A" SG Safety Valve indicates open.
3. The crew transitions to ES-1.1, *SI Termination*, with conditions as follows:
   - Charging pumps: "A" and "B" running
   - RCS pressure: 1800 psia and stable
   - Pressurizer level: 22% and stable
   - RCS subcooling: 90°F and increasing.
   - Containment temperature: 145°F and stable

Per ES-1.1, the crew stops the "A" Charging Pump.

Assuming the LOCA plus the partially open Safety Valve result in RCS pressure steadily decreasing after stopping the CHS Pump, what action is the US required to take per ES-1.1?

a) Leave the "A" Charging Pump off and transition to ES-1.2, *Post LOCA Cooldown and Depressurization.*
b) Restart the "A" Charging Pump and transition to E-1, *Loss or Reactor or Secondary Coolant.*
c) Remain in ES-1.1, *SI Termination*, and restart the "A" Charging Pump if Pressurizer level decreases below SI Reinitiation Criterion.
d) Remain in ES-1.1, *SI Termination*, and restart the "A" Charging Pump if RCS subcooling level decreases below SI Reinitiation Criterion.

Proposed Answer: A

Explanation (Optional): This question is considered SRO since it requires the applicant to assess plant conditions and determine that a transition to an event-specific Emergency Sub (ES) Procedure is required. Also, the applicant must apply EOP rules of usage and determine whether or not foldout page criteria apply at this specific point in the EOP. This question is considered a KA match, since a small LOCA is in progress, and a steam release path exists through a SG safety valve. Also, the steam release is interfering with SI Termination by causing a cooldown to continue (subcooling increasing), causing RCS pressure to continue to decrease after the Charging Pump is stopped. In order to mitigate the effects of the steam release, the crew is required to transition from ES-1.1 to ES-1.2. The US will transition to ES-1.2, since RCS pressure continuing to decrease indicates SI termination will not be successful ("C" and "D" wrong). The US will not restart the second Charging Pump, since SI Reinitiation Criteria does not apply until SI has been terminated ("A" correct, and "B" wrong). Also, the proper transition is to ES-1.2 at this point, not E-1. "B" is plausible, since if SI reinitiation criteria were in effect, the crew would be required to restart ECCS pumps as necessary, and after SI has been terminated, transition to E-1 is directed if conditions degrade. If the crew were to restart the charging pump at this point and transition to E-1, RCS conditions would again stabilize, and after transitioning to E-1, the crew would again transition to ES-1.1 per E-1, step 6, and be stuck in a "do-loop". If the crew restarted the Charging Pump and transitioned to ES-1,2, this procedure would re-stop the pump after checking subcooling. After transitioning to ES-1.2, the crew will commence a cooldown to Cold Shutdown, which increases subcooling, allowing an operator controlled depressurization of the RCS, which will increase injection flow and decrease break flow, eventually allowing SI to be terminated. "C" and "D" are plausible, since subcooling and Pzr level comprise the ES-1.1 SI Reinitiation Criteria.

Technical Reference(s): ES-1.1 (Rev. 16-2), steps 3 and 4

(Attach if not previously provided, ES-1.1 (Rev. 16-2), Foldout Page SI Reinitiation Criteria

including version/revision number) OP 3272 (Rev. 09-0), Section 1.6.

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-05528 Identify plant conditions that require entry into EOP 35 ES-1.2. (As available)

Question Source: Bank #74743

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5 and 43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 89	Tier #		2
K/A Statement: Predict impact/mitigate consequences of	Group #		1
an AC Electrical Distribution System load, that if	K/A #	062.A2.01	
deenergized, would degrade/hinder plant operations	Importance Rating		3.9

Proposed Question:

1. A small break LOCA occurs.
2. MCC 32-2R deenergizes.
3. The crew transitions to ES-1.2, *Post LOCA Cooldown and Depressurization.*
4. The crew is currently at ES-1.2, step 21 "Verify ECCS Flow Not Required", and the following conditions exist:
   - Pressurizer Level: 20% and slowly increasing.
   - Core Exit TCs: 460°F and slowly decreasing.
   - RCS pressure: 785 psia and stable.
   - CTMT temperature: 145°F and stable.
   - MCC 32-2R: Still deenergized.

Per ES-1.2, is the US is required to direct the crew to start ECCS Pumps? If so, why? If not, what action(s) is the US required to direct that will prevent with the "A" Train SI Accumulators from injecting as the RCS is depressurized?

a) Starting ECCS Pumps IS required, since inadequate subcooling exists.
b) Starting ECCS Pumps IS required, since inadequate Pressurizer level exists.
c) ECCS flow is NOT required. Using GA-7, isolate the SI Accumulators.
d) ECCS flow is NOT required. Using GA-7, vent nitrogen off the SI Accumulators.

Proposed Answer: D

Explanation (Optional): This question is considered SRO since it requires the applicant to assess plant conditions, and determine whether ECCS flow is required, which is beyond system knowledge. It also requires determining specific actions required per the GA procedure with a loss of electrical bus. "A" is wrong, since subcooling is 57°F, which is above the required 32°F. "A" is plausible, since 57°F is below the minimum subcooling required if Containment were adverse, and this action is required if inadequate subcooling exits. Also, Containment temperature is elevated. "B" is wrong, since Pzr level is above the minimum required level of 16%. "B" is plausible, since Pzr level is below that required if Containment were adverse, and this action is required if inadequate PZR level exists. Also, Containment temperature is elevated. "C" is wrong, but plausible, since this action is required per the GA if MCC 32-2R were available. "D" is correct, since adequate subcooling exists (57°F), and adequate PZR level exists (38%) so that SI is not required, and accumulator injection is neither required nor desired. As the plant is being cooled down to cold shutdown, RCS pressure will drop below accumulator injection setpoint. Accumulators are normally isolated at this point to prevent N2 injection into the RCS. "A" and "C" Accumulator isolation valves are normally open MOVs, powered from "A" Train 480V MCC 32-2R, so they cannot be closed due to loss of power . GA-7 directs the crew to vent the unisolable accumulators, and this is possible, since the vent valves are in parallel for each accumulator, powered from opposite train 125VDC power ("D" correct).

Technical Reference(s): (Attach if not previously provided, including version/revision number)	ES-1.2 (Rev. 20-0), Steps 21-23 EOP 35 GA-7 (Rev. 00-1), steps 1 and 3.RNO OP 3310B-004 (Rev. 05-0), pages 2 and 3 of 3 P &ID 112B (Rev. 23)
Proposed references to be provided to applicants during examination:	Steam Tables
Learning Objective:	MC-07428 Given a set of plant conditions, determine the required actions to be taken per ES-1.2. (As available)
Question Source:	New
Question History:	
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55. 43.5
Comments:	

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 90	Tier #		2
K/A Statement: Knowledge of the operational	Group #		1
implications of EOP warnings, cautions, and	K/A #	063.GEN.2.4.20	
notes for the DC Electrical Distribution System	Importance Rating		4.3

Proposed Question:

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

1. The crew enters EOP 3509.1, *Control Room, Cable Spreading Area or Instrument Rack Room Fire*.
2. Control power fuses supplying two "A" Train Accumulator Vent Supply Valves, 3SIL*SV8875A and 3SIL*SV8857B, blow due to the fire.
3. The crew evacuates the Control Room.

Complete the following statements concerning when during the performance of EOP 3509.1 the blown fuses will become evident, and where in the procedure fuse replacement will be directed to address the loss of Control Power.

The blown control power fuses will become evident by a loss of indicating lights when __(1)__. Fuse replacement will be directed __(2)__.

a) (1) realigning the Umbilical Cord at the Auxiliary Shutdown Panel
   (2) at the appropriate procedure step in the body of EOP 3509.1
b) (1) realigning the Umbilical Cord at the Auxiliary Shutdown Panel
   (2) per Attachment A, "Primary Side PEO Actions on a Control Room Evacuation"
c) (1) placing the transfer switches in LOCAL at the Transfer Switch Panel
   (2) at the appropriate procedure step in the body of EOP 3509.1
d) (1) placing the transfer switches in LOCAL at the Transfer Switch Panel
   (2) per Attachment A, "Primary Side PEO Actions on a Control Room Evacuation"

Proposed Answer: C

Explanation (Optional): This question is considered SRO, since it requires the applicant to apply an EOP note to a specific situation, and determine whether to use the body of the procedure or an Attachment to address the situation. Fire-related shorts can collectively draw excessive current and blow control power fuses, which provide indication and control. This will become evident when the component is placed in LOCAL at the Transfer Switch Panel. "A" and "B" are wrong, since realigning the umbilical cord at the Aux Shutdown Panel affects indication, but does not affect Accumulator valve position indication. "A" and "B" are plausible, since other indication is affected by this action, and this action is directed along with the Transfer Switches in Attachment E. "C" is correct, and "D" wrong, since the body of the procedure directs fuse replacement. "D" is plausible, since numerous local actions are being taken by the PEOs per Attachments "A" and "B" to address fire-related issues.

Technical Reference(s): EOP 3509.1 (Rev. 20-0), Attachment E, Note prior to step 1, and step 4
(Attach if not previously provided, EOP 3509.1 (Rev. 20-0), step 30.d.
including version/revision number) EOP 3509.1 (Rev. 20-0), Attachment A
EOP 3509.1 basis document (Rev. 19-0), for Att. E, steps 1 and 4.

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06189 Discuss the basis of major EOP 3509.1 procedure steps and/or sequence of steps (As available)

Question Source: New

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 91	Tier #		2
K/A Statement: Knowledge of the events related to Hydrogen	Group #		2
Recombiner and Purge Control operation/status that must be	K/A #	028.GEN.2.4.30	
reported to internal organizations or outside agencies	Importance Rating		4.1

Proposed Question:

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

Time: Event:

0000: A mechanic informs the Control Room that while constructing scaffolding in the area of the Containment Purge System, scaffolding piping inadvertently struck a valve in the area.
0005: A PEO is sent to investigate, and reports that the handwheel, lock, and chain are broken off of CTMT Purge Inlet Isolation valve, 3HVU*CTV32A.
0005: The crew logs into LCO 3.6.1.7, "CONTAINMENT VENTILATION SYSTEM."
0105: Maintenance reports that repairs to 3HVU*CTV32A are required, and that the valve can NOT be relocked until repairs are complete.
0305: The crew commences a Plant Shutdown as required by LCO 3.6.1.7, "CONTAINMENT VENTILATION SYSTEM."
0452: The crew manually trips the reactor as part of the shutdown process.
0459: Safety Injection automatically actuates due to loss of pressure control on the reactor trip.

What was the first event that required a NRC notification to be made per RAC 14, *Non-Emergency Station Events*?

a) The discovery of the broken lock.
b) The initiation of the plant shutdown.
c) The manual actuation of the Reactor Protection System.
d) The ECCS discharge into the Reactor Coolant System.

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it involves knowledge of administrative procedures (Reportability) specifically related to the SRO job function. Logging into an LCO requiring a plant shutdown is not reportable ("A" wrong). "B" is correct, and "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 prior to the other events in the timeline. "A" is plausible, since this LCO requires a shutdown, and commencing the shutdown is reportable. "C" and "D" are plausible, since these events also require a 4 hour report.

Technical Reference(s): RAC 14 (Rev. 011-00), Attachment 1, Sheets 1 and 3 of 4.
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-00016 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 History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 92	Tier #		2
K/A Statement: Predict impact/mitigate loss of one, two or	Group #		2
three charging pumps on the Pressurizer Level Control System	K/A #	011.A2.4	
Proposed Question:	Importance Rating		3.7

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

1. RO notices the "A" CHS Pump amps are oscillating.
2. Before the operators can take action, the "A" CHS Pump trips.
3. The RO isolates charging and letdown.
4. The crew enters AOP 3580, *Loss of All Charging Pumps*.
5. The RO verifies a suction path to the Charging Pumps is aligned from the VCT.
6. The STA reports RCP #1 seal inlet temperatures for all four RCPs indicate 140°F.

Per AOP 3580, what are ALL of the actions required prior to starting the "B" Charging Pump?

a) Trip the reactor, trip all four Reactor Coolant Pumps (RCPs), and enter E-0, *Reactor Trip or Safety Injection*.
b) Close the Seal Return (CBO) Flow Isolation Valves.
c) Vent the "B" CHS Pump gravity feed boration line.
d) Vent the "B" CHS Pump gravity feed boration line and close the Seal Return (CBO) Flow Isolation Valves.

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions, and determine specific procedural actions to be taken per the applicable AOP.
Per AOP 3580, step 6, if if seal inlet temperatures are greater than 190°F, the crew will be required to trip the reactor, trip the affected RCPs, and go to E-0. "A" is wrong, since seal inlet temperatures are less than 190°F. "A" is plausible, since this would be required if temperature was above 190°, and the foldout page also has requirements to trip the reactor. With seal inlet temperatures less than 190°F, step 6.g, will direct the crew to check seal cooling, and if no seal cooling is aligned, the crew will close the affected RCP CVC Seal Return (CBO) Flow Isolation Valves. "B" and "D" are plausible, since the normal method of seal cooling is seal injection flow, and this has been lost. "B" is wrong, since thermal barrier cooling is still being supplied by RPCCW to the thermal barrier heat exchangers. "C" is correct, and "D" plausible, since, per step 5, the initial oscillation of Charging pump amps requires the crew to vent the gravity feed boration line prior to step 7.d, which directs the crew to start one charging Pump.

Technical Reference(s): AOP 3580 (Rev. 03-0), steps 1-7.
(Attach if not previously provided, AOP 3580 (Rev. 03-0), Foldout Page
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07667 Given a set of plant conditions, determine the required actions to be taken per AOP 3580. (As available)

Question Source: Bank #76155

Question History: Millstone 3 2011 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 # 93	Tier #		2
K/A Statement: Predict impact/mitigate a	Group #		2
misaligned rod in the Rod Control System	K/A #	014.A2.4	
Proposed Question:	Importance Rating		3.9

With the plant at 80% power with an up-power in progress, the following sequence of events occurs:

1. The RO reports Control Bank "D" group 1 rod M-12 indicates 15 steps below the rest of Control Bank "D".
2. The crew enters AOP 3552, *Malfunction of the Rod Drive System.*
3. The crew proceeds to AOP 3552, Attachment A, "Misaligned Rod".
4. I&C reports a blown stationary gripper coil fuse for rod M-12.
5. I&C reports replacement fuse is available.
6. The crew performs a Shutdown Margin Calculation.
7. The rod has been misaligned for 20 minutes.

Which of the choices below describes action(s) that is/are required to be taken by the US per AOP 3552?

a) Shutdown the plant, being in MODE 3 within 6 hours.
b) Realign the Bank to the inoperable rod within 1 hour, and restore the rod to operable within 72 hours.
c) Reduce power to <75% power, and lower the NIS high Flux Trip setpoint to 85% within 2 hours.
d) Realign the rod. Then, using Attachment E, restore the P-to-A Converter.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions, and determine specific AOP action to take to mitigate the event, and apply Tech Spec ACTIONS (below the line). This question is related to the partially dropped, stuck rod at Farley, Unit 2 on 10/14/2002. During rod testing, rod became stuck after partially inserting into the core, and procedures were not adequate at addressing the rod. With the stationary gripper coil fuse blown, the rod has no power and is stuck. "A" is correct, and "B" wrong, since with an untrippable rod, AOP 3552 directs the crew to shutdown the plant. "B" is plausible, since this action would be appropriate for more than one misaligned rod if it were not stuck. "C" is wrong, but plausible, since this action would be appropriate per Tech Specs if the rod was misaligned, but trippable. "D" is wrong, since the rod is stuck. "D" is plausible, since AOP 3552, step 6 directs the crew to proceed to Attachment A, and align the rod. This action would be correct per AOP 3552 if the rod was misaligned, but trippable.

Technical Reference(s):	AOP 3552 (Rev. 14-0), step 6
(Attach if not previously provided,	AOP 3552 (Rev. 14-0), Att A, steps A.1.c, o, A.4.b, step A.6.b and e
including version/revision number)	AOP 3552 (Rev. 14-0), Att D, Note prior to step D.3 and D.5
	Tech Spec LCO 3.1.3.1 (Amendments 60 and 258)

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-03901 Describe the major action categories contained within AOP 3552.	(As available)

Question Source: New

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.2

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 94	Tier #		3
K/A Statement: Ability to use procedures related to shift	Group #		1
staffing, minimum crew compliment, overtime limitations, etc.	K/A #	GEN.2.1.5	
Proposed Question:	Importance Rating		3.9

Initial Conditions:

- Shift turnover is in progress.
- One oncoming PEO called in sick, and one of the off-going PEOs volunteers to work 4 hours of overtime to assist with the load increase.
- The current day and time is Sunday night at 1800.

The off-going PEO's work history is as follows:

- Monday: 0700-1500
- Tuesday: 0600-1800
- Wednesday: Off
- Thursday: 0600-1800
- Friday: 0600-1800
- Saturday: 0600-1800
- Sunday: 0600-1800

Complete the following statement.

The PEO is NOT eligible to work the 4 hours of overtime without additional authorization since this would violate the maximum__________requirement.

a) consecutive hours worked
b) hours worked in a 24-hour period
c) hours worked in a 48-hour period
d) hours worked in a 7-day period

Proposed Answer: C

Explanation (Optional): Section 3.3.1 of LI-AA-700 states the "cognizant supervisor" is responsible for applying work hour limits. Therefore, this question is considered SRO level. "A" is wrong, since the PEO has not exceeded the maximum consecutive hours limit of 16 hours. "B" is wrong, since the PEO has not exceeded the maximum of 16 hours in a 24-hour period. "C" is correct, since the PEO would exceed the maximum of 26 hours in any 48-hour period, since for the period from Saturday at 0600 until Monday at 0600; the PEO has already worked 24 hours. "D" is wrong, since the PEO has worked 68 hours in the past 7 days, which is less than the maximum-allowed 72 hours in a 7-day period. "A", "B", and "D" are plausible, since the PEO has been working a large number of hours, and there are limits for each of these cases.

Technical Reference(s): LI-AA-700 (Rev. 13-0), section 3.3
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06863 State the overtime limits for Millstone Personnel (As available)

Question Source: Bank #80918

Question History: Millstone 3 2007 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 95	Tier #		3
K/A Statement: Ability to determine	Group #		2
Technical Specification Mode of Operation	K/A #	GEN.2.2.35	
Proposed Question:	Importance Rating		4.5

The plant is initially in MODE 6, with a full core offload in progress per OP3210B, *Refueling Operations*.

Complete the following statement.

Per Tech Spec Bases, the point in the core offload when MODE Zero is entered is when the last fuel assembly has been________.

a) removed from the reactor vessel
b) transferred out of Containment and has cleared the transfer canal
c) transferred out of Containment and the Fuel Transfer Tube Isolation Valve is closed
d) placed into its required position in one of the Spent Fuel Pool storage racks

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it involves administrative requirements for determining MODE changes during refueling/fuel handling operations, which requires knowledge of Tech Spec Bases. The transition from MODE 6 to MODE ZERO is procedurally defined per OP 3210B, when all fuel assemblies have been removed from CTMT to the Spent Fuel Pool. This is specifically defined in Tech Spec Bases to occur when "the last fuel assembly of a full core offload has been transferred to the Spent Fuel Pool ("A" wrong) and has cleared the transfer canal ("B" correct) while in transit to a storage location" ("C" and "D" wrong). Knowledge of Tech Spec Basis information ensures MODE 6 requirements are not relaxed prematurely during fuel movement in Containment. "A" is plausible, since all fuel assemblies have been removed from the Reactor Vessel while transitioning from MODE 6 to MODE Zero. "C" and "D" are plausible, since all fuel bundles will be placed in the spent fuel pool racks after entering MODE Zero, and the Gate Valve can be closed to isolate Containment from the Spent Fuel Pool.

Technical Reference(s): OP 3210B (Rev. 010-04), Note Prior to step 4.1.13
(Attach if not previously provided, Tech Spec Basis 3/4.9.1 (Amendment #230)
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-06388 Describe the basis for major procedural steps and/or sequence of steps in OP 3210 series (As available)

Question Source: Bank #69779

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2, 43.6 and 43.7

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 96	Tier #		3
K/A Statement: Knowledge of the process	Group #		2
for conducting special or infrequent tests	K/A #	GEN.2.2.7	
Proposed Question:	Importance Rating		3.6

Initial conditions:

- The crew commences a Reactor startup per OP 3202, *Reactor Startup*, which is designated as an ICCE (Infrequently Conducted or Complex Evolution).
- Source Range countrate is 10 cps.

Current conditions:

- Source Range countrate is 100 cps.
- The value of the 1/M plot is "0.4."
- The Reactor Engineer reports the projected critical position based on the 1/M plot is Control Bank C at 45 steps.

Is the crew allowed to proceed with the startup? If not, what action is the US required to direct?

a) Yes. The crew is allowed to proceed with the reactor startup.
b) No. Maintain rods at the present rod height and recalculate the Estimated Critical Condition.
c) No. Initiate immediate boration and drive all control bank rods into the core.
d) No. Trip the reactor.

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it involves knowledge of administrative procedures (ICCE) specifically related to the SRO job function, and requires the applicant to assess plant conditions and determine a specific transition requirement during a reactor startup. OP 3202 is designated as an ICCE. ICCEs require criteria for terminating the test or evolution. One of the identified Termination Criteria for a reactor startup is criticality predicted below the RIL. Per Table in the COLR, RIL for zero percent power is Bank C at 51 steps, which is above what the 1/M plot is predicting. After three doublings, any time calculations or instruments indicate that the reactor may go critical below the RIL, reactor startup must be terminated, all control banks inserted and immediate boration performed ("C" correct, "A", "B", and "D" wrong). "A" is plausible, since this would be correct if 1/M were above RIL. "B" is plausible, since ECC is required to be recalculated when criticality is predicted to be outside of the admin reactivity band. "D" is plausible, since termination criteria exist in OP 3202 that require a reactor trip, including a sustained SUR of 1.0 dpm, or an uncontrolled cooldown resulting in Tc being less than 530°F.

Technical Reference(s): OP-AA-106 (Rev. 10-0), step 3.2.2.c.8
(Attach if not previously provided, OP 3202 (Rev. 23-0), Cover Sheet
including version/revision number) OP 3202 (Rev. 23-0), Section 3.14 and step 4.25.1 and 4.25.2

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-03375 Discuss conditions which require transition to other procedures from OP 3202. (As available)

Question Source: Bank #60751
Question History:
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.1, 41.6, and 43.5
Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 97	Tier #		3
K/A Statement: Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Group #		3
	K/A #	GEN.2.3.15	
	Importance Rating		3.1

Proposed Question:

Initial conditions:

- The plant is at 100% power.
- N-16 Radiation Monitor, MSS-RE80A, is NOT in service.

The following sequence of events occurs:

1. Air Ejector Rad Monitor ARC21-1 goes into ALERT.
2. The crew enters AOP 3576, *Steam Generator Tube Leak.*
3. The RO reports Blowdown Rad Monitor 3SSR08-1 is trending upward.
3. Chemistry is dispatched to sample all 4 SGs for activity.
4. The crew is currently at AOP 3576, step 3 "Check Primary to Secondary Leakage," and current conditions are as follows:
   - Chemistry reports their initial sample results should be available in about 15 minutes.
   - The crew anticipates that SSR08 will exceed the ALERT setpoint in about five minutes.

Complete the following statement concerning verification of leakage per OP 3272, *EOP Users Guide*, and AOP 3576, and the procedural action the US is required to direct.

Leakage __(1)__ verified. The US is required to __(2)__.

a) (1) IS
   (2) continue on to AOP 3576, step 4
b) (1) IS
   (2) go to OP 3204, *At Power Operations*, and commence a shutdown to MODE 3 within 6 hours
c) (1) IS NOT
   (2) wait for a Chemistry sample to confirm the presence of leakage
d) (1) IS NOT
   (2) wait to see if SSR08-1 exceeds the ALERT setpoint to confirm the presence leakage

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it requires the applicant to use Radiation Monitor data to make a decision as to whether or not to move forward in an AOP. This also involves knowledge of EOP Users' Guide guidance on interpreting Rad Monitor trends that are increasing but not in an alarm condition. To progress in AOP 3576 with the N-16 monitors out of service, two indications of tube leakage are required. ARC21 Rad Alert AND SSR08 trending upward satisfy the requirement ("A" correct, "C" and "D" wrong) even without the chemistry sample results. "C" and "D" are plausible, since the N-16 monitors are out of service, and SSR08 has not yet reached the ALERT setpoint. "B" is wrong, since shutting down the plant is only required if leakage is determined to exceed 150 gallons per day. "B" is plausible, since leakage is confirmed, and a shutdown may be required, based on leak rate.

Technical Reference(s): (Attach if not previously provided, including version/revision number)	AOP 3576 (Rev. 08-0), steps 2 and 3.a-d. OP 3272 (Rev. 09-0), Attachment 5, Definition of "Normal".

Proposed references to be provided to applicants during examination: None

Learning Objective: MC-07573 Given a set of plant conditions, determine the required actions to be taken per AOP 3576. (As available)

Question Source: Bank #72474

Question History:

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.11 and 43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 98	Tier #		3
K/A Statement: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	Group #		3
	K/A #	GEN.2.3.5	
	Importance Rating		2.9

Proposed Question:

A LOCA occurs, and Initial Conditions are as follows:

- An ALERT-Charlie 1 has been declared at Millstone 3.
- The CR-DSEO is preparing the Incident Report Form.
- The CR-DSEO is currently determining if the "Radiological Release in progress due to event" box is required to be checked on the IRF form.

The CR-DSEO gathers the following information from the Radiation Monitor Computer:

1. ESF Bldg Normal Vent Exhaust Monitor 3HVQ-RE49 shows an increasing trend, but is NOT in ALERT.
2. Turbine Building Stack Monitor 3HVR-RE10A is in ALERT, but is NOT in ALARM.
3. Liquid Waste Effluent Monitor 3LWS-RE70 is in ALERT, but is NOT in ALARM.
4. Containment Area (Fuel Drop) Radiation Monitor 3RMS-RE41 is in ALARM.

In accordance with MP-26-EPI-FAP06, *Classification and PARs*, which one of these Radiation Monitors specifically requires the CR-DSEO to select the "Radiological release in progress due to event" Box?

a) 3HVQ-RE49 increasing trend
b) 3HVR-RE10A in ALERT
c) 3LWS-RE70 in ALERT
d) 3RMS-RE41in ALARM

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it involves knowledge of administrative procedures related to the SRO job function in the Emergency Plan. It requires the applicant to interpret radiation readings as they pertain to making an appropriate decision about whether a radiation release considered to be in progress during an event. The Radiation Monitor input to checking the block "Radiological release in progress due to event" is a gaseous effluent radiation monitor ("A" plausible) in ALERT ("B" correct, "A" wrong) or ALARM ("C" and "D" plausible). "C" and "D" are wrong, since these are not gaseous effluent monitors. "A" is plausible, since it is a gaseous effluent monitor, and an increasing trend is considered "not normal" when diagnosing RMS Alarm/Alert status if it is anticipated it will reach the ALERT setpoint. "C" is plausible, since this monitors an effluent point, and RE10A is only in ALERT. "D" is plausible, since this monitor is in ALARM, and is used on the EAL tables to determine classification level.

Technical Reference(s):	MP-26-EPI-FAP06 (Rev. 10-0), Section 2.1.8
(Attach if not previously provided,	MP-26-EPI-FAP06 (Rev. 10-0), Att. 6
including version/revision number)	

Proposed references to be provided to applicants during examination: None

Learning Objective:	MC-02534 The Shift Manager and Unit Supervisor will perform all administrative actions necessary to protect the public in accordance with emergency plan procedures.	(As available)

Question Source:	Bank #89303
Question History:	Millstone 3 2009 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	55.41.11, 43.4, and 43.5
Comments:	

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 99	Tier #		3
K/A Statement: Knowledge of the	Group #		4
emergency plan	K/A #	GEN.2.4.29	
Proposed Question:	Importance Rating		4.4

The Control Room DSEO has just declared a General Emergency - BRAVO.

How will the CRDSEO initially notify the state of the PAR, and what evacuation, if any, will be carried out by the state?

a) The Incident Report Form will serve as the PAR notification. State officials will evacuate a 2-mile radius around the site.
b) The Incident Report Form will serve as the PAR notification. An evacuation will NOT be conducted for a General Emergency BRAVO.
c) 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.
d) 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.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it involves knowledge of Emergency Plan procedures specifically related to the SRO job function. 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" correct, "C" wrong). "C" and "D" 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. 06-0), Note prior to Section A
(Attach if not previously provided,
including version/revision number)

Proposed references to be provided to applicants during examination:

Learning Objective: EP-00203 Explain the method for providing protective action recommendations initially and following activation of the Emergency Response Organization. (As available)

Question Source: Bank #85285

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.12 and 55.43.5

Comments:

Examination Outline Cross-Reference:	Level	RO	SRO
Question # 100	Tier #		3
K/A Statement: Knowledge of SRO responsibilities in emergency plan implementation	Group #		4
	K/A #	GEN.2.4.40	
Proposed Question:	Importance Rating		4.5

A "General Emergency" has been declared.

Which of the following tasks can NOT be re-assigned by the Control Room - Director of Station Emergency Organization (CR-DSEO) to other available Control Room individuals?

a) Conducting the station evacuation.
b) Notifying the NRC of a 10CFR50.54(x) invocation via ENS.
c) Announcing termination of sheltering.
d) Issuing KI tablets to the control room staff.

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it involves knowledge of Emergency Plan procedures specifically related to the SRO job function. "A" is correct, since the CRDSEO cannot delegate issuance of KI tablets. "B", "C", and "D" are wrong since these tasks cannot be delegated. "B", "C", and "D" are plausible since these are all responsibilities of the CR DSEO, and all four choices are implementation actions, not important managerial decisions, such as developing a PAR.

Technical Reference(s): MP-26-EPI-FAP01-001 (Rev. 14-0), Section E. 10
(Attach if not previously provided, MP-26-EPI-FAP01-001 (Rev. 14-0), Section F, steps 12, 14, and 16
including version/revision number)

Proposed references to be provided to applicants during examination: None

Learning Objective: EP-00208 List the responsibilities of the Shift Manager while serving as CR Director of Station Emergency Operations (CR DSEO). (As available)

Question Source: Bank #75654

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:

Millstone 3 2017 NRC Initial/Upgrade Written Exam Answer Sheet

NAME: Key DATE: 12/11/17

QUES	ANSWER CHOICE				QUES	ANSWER CHOICE			
1	A	B	●	D	26	A	●	C	D
2	A	●	C	D	27	A	B	C	●
3	●	B	C	D	28	A	●	C	D
4	A	B	●	D	29	A	B	●	D
5	A	B	●	D	30	A	B	C	●
6	A	B	C	●	31	A	●	C	D
7	A	B	●	D	32	●	B	C	D
8	●	B	C	D	33	A	●	C	D
9	●	B	C	D	34	A	●	C	D
10	A	●	C	D	35	A	B	●	D
11	●	B	C	D	36	A	B	●	D
12	A	●	C	D	37	A	●	C	D
13	●	B	C	D	38	A	●	C	D
14	A	B	●	D	39	●	B	C	D
15	A	B	C	●	40	A	●	C	D
16	A	B	●	D	41	A	B	C	●
17	●	B	C	D	42	A	B	C	●
18	A	B	C	●	43	A	●	C	D
19	A	B	●	D	44	A	B	C	●
20	A	B	C	●	45	A	B	●	D
21	●	B	C	D	46	A	B	●	D
22	A	B	C	●	47	A	B	C	●
23	A	B	●	D	48	●	B	C	D
24	A	●	C	D	49	A	B	C	●
25	●	B	C	D	50	A	B	C	●

Millstone 3 2017 NRC Initial/Upgrade Written Exam Answer Sheet

NAME: Key   DATE: 12/11/17

QUES	ANSWER CHOICE				QUES	ANSWER CHOICE			
51	A	B	C	●	76	●	B	C	D
52	A	B	C	●	77	A	B	●	D
53	A	B	●	D	78	A	B	C	●
54	A	●	C	D	79	A	●	C	D
55	A	B	●	D	80	A	B	C	●
56	A	B	C	●	81	●	B	C	D
57	A	●	C	D	82	A	B	●	D
58	●	B	C	D	83	A	B	●	D
59	A	B	●	D	84	A	●	C	D
60	●	B	C	D	85	A	B	C	●
61	A	B	●	D	86	A	B	C	●
62	A	●	C	D	87	A	●	C	D
63	●	B	C	D	88	●	B	C	D
64	●	B	C	D	89	A	B	C	●
65	A	B	C	●	90	A	B	●	D
66	A	B	C	●	91	A	●	C	D
67	A	B	●	D	92	A	B	●	D
68	●	B	C	D	93	●	B	C	D
69	A	B	●	D	94	A	B	●	D
70	●	B	C	D	95	A	●	C	D
71	●	B	C	D	96	A	B	●	D
72	●	B	C	D	97	●	B	C	D
73	A	●	C	D	98	A	●	C	D
74	A	●	C	D	99	●	B	C	D
75	A	●	C	D	100	A	B	C	●