Examination Outline Cross-reference:	Level	RO	SRO
Question # 1	Tier #	1	1
K/A Statement: Operational Implications of	Group #	1	1
decay 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 due to a loss of offsite power.
2. The crew enters ES-0.1, *Reactor Trip Response*.
3. One hour after the trip, the BOP operator reports 380 gpm of Auxiliary Feedwater flow is required to maintain RCS Tcold stable at 557°F.

Considering only decay heat, how long after the trip will 90 gpm of AFW flow first be adequate to maintain RCS temperature stable?

a) 2 to 4 hours
b) 8 to 12 hours
c) 1 to 2 days
d) 6 to 8 days

Proposed Answer: D

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 major 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, and 95 gpm one week after the trip.
"D" is correct, since 90 gpm flow is not adequate until after one week, where approximately 95 gpm is required. This is confirmed by ECA-1.1, Attachment 1, which shows actual required ECCS flow during a LOCA to remove decay heat after a trip. Per ECA-1.1, Attachment 1, after one week, (10,080 hours) about 90 gpm is required.
"A" is wrong, since after 4 hours, approximately 250 gpm is required.
"B" is wrong, since after 12 hours, approximately 190 gpm is required.
"C" is wrong, since after 2 days (2880 hours), approximately 160 gpm is required.
"A", "B", and "C" are plausible, since the question provides the flow required one hour after the trip, and each of the distractors involve time periods beyond one hour, where decay heat levels will be lower.

Technical Reference(s): ECA-1.1 (Rev. 19), Attachment A
(Attach if not previously provided, including version/revision number.) General Physics Reactor Theory Text (Rev. 4) Chapter 8, Fig. 8-26
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major administrative and procedural precautions and limitations place on the operation of the Auxiliary Feedwater System, and the basis for each
Question Source: Modified Bank #409303 (Parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.1, 41.8, 41.10, 41.14

Comments:
The question is considered “modified” since the amount of flow required in the question has changed, creating a new correct answer.

Original Bank Question

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

Answer: C

Examination Outline Cross-reference:	Level	RO	SRO
Question # 2	Tier #	1	1
K/A Statement: Operational Implications of	Group #	1	1
Thermodynamics and flow characteristics of open or	K/A #	APE.008.AK1.1	
leaking valves during a Pzr vapor space accident	Importance Rating	3.2	3.7

Proposed Question:

Current conditions:

- The crew is recovering from a faulted SG outside CTMT, upstream of the MSIVs.
- SI has been terminated.
- The crew is maintaining the plant in a stable condition per ES-1.1, *SI Termination*.
- PZR pressure is 2300 psia and stable.
- PZR level is 60% and stable.
- PRT pressure is 30 psia.

A significant leak occurs through one of the PZR Safety valves, causing RCS pressure to rapidly decrease.

Based on the event in progress, complete the following statement.

The parameter that will require the crew to manually reinitiate Safety Injection is low___(1)___; and the temperature of the steam in the tailpipe downstream of the Pzr Safety valve is approximately___(2)___.

	(1)	(2)
a)	PZR level	250°F
b)	PZR level	650°F
c)	RCS subcooling	250°F
d)	RCS subcooling	650°F

Proposed Answer: C

Explanation (Optional): On a vapor space break, Pzr level is not a valid indication of RCS inventory, since pressure will quickly drop to saturation in the vessel and the hot legs, causing a two phase mixture that will expand and force flow up the surge line and into the Pzr, preventing Pzr level from dropping as it normally does on a LOCA. This causes Pzr level to increase until the Pzr is full. The enthalpy of the saturated fluid in the vapor space does not change as it passes through a safety valve, resulting in a temperature indication corresponding to saturation temperature of the pressure downstream of the valve, which is at PRT pressure.

"A" and "B" are wrong, since Pzr level will not drop significantly on a vapor space break. "A" and "B" are plausible, since there is a Pzr low level SI reinitiation criterion, and for most small break LOCAs, PZR level will decrease, and this would be the parameter that requires SIS reinitiation.

"C" is correct, since the RCS low subcooling SI reinitiation criterion of 32°F will be met, and the Pzr low level criterion will not be met. And, per the steam tables, saturation temperature for 30 psia is 250°F.

"D" is wrong, since the enthalpy of the saturated fluid in the vapor space does not change as it passes through a safety valve, resulting in a temperature indication corresponding to the pressure downstream of the valve, which is at PRT pressure. Per the steam tables, saturation temperature for 30 psia is 250°F. "D" is plausible, since 655°F is saturation temperature for Pzr pressure.

Technical Reference(s): ES-1.1 (Rev. 17), Foldout Page SI Re-initiation Criteria
(Attach if not previously provided, Westinghouse MITCORE text pg 16-45 and 16-46
including version/revision number.) Steam Tables
Proposed references to be provided to applicants during examination: **Steam tables**
Learning Objective: Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of ES-1.1.
Question Source: Bank #407947
Question History: Last NRC Exam Millstone 3 2002 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.8 and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 3	Tier #	1	1
K/A Statement: Ability to operate or monitor	Group #	1	1
the RCS temperature detection subsystem	K/A #	APE.015.AA1.9	
during a RCP malfunction	Importance Rating	3.1	3.2

Proposed Question:

With Reactor power initially at 25%, the following sequence of events occurs:

1. The "D" RCP trips.
2. The crew enters AOP 3554, *RCP Trip or Stopping an RCP at Power*.
3. The RO monitors RCS temperature for proper response at MB4.

Assuming the reactor does NOT trip and no operator action has been taken, complete the statement below.

"D" RCS Loop temperature will go to_____.

a) Tsat for the "D" SG
b) Tave of the other 3 loops
c) Thot of the other 3 loops
d) Tcold of the other 3 loops

Proposed Answer: D

Explanation (Optional):
"D" is correct, and "A", "C", and "D" wrong, since when the "D" RCP stops, forced circulation stops in that loop. Since the other RCPs keep cold leg (vessel inlet) pressure high, and there is a pressure drop across the core, the driving head for the "D" RCS loop is now the DP across the reactor vessel, which causes reverse flow in the loop. So Tcold water starts reverse-flowing into the "D" loop. The affected SG will steam significantly less than the other loops, since less energy is being added to it from the RCS. So its differential temperature is low, and the entire loop approaches Tcold of the other loops.
"A" is plausible, since during natural circulation conditions when no RCPs are running, Tcold for each of the RCS loops goes to Tsat for the SGs.
"B" is plausible, since this is the normal average temperature of a RCS loop with all RCPs running.
"C" is plausible, since this occurs if heat sink is not adequate in a single RCS loop.

Technical Reference(s): AOP 3554 (Rev. 11), steps 1, 4, and 6
(Attach if not previously provided, AOP 3554 Basis Doc (Rev. 11), steps 6 and 7
including version/revision number.) P&ID 102A (Rev. 34)
Proposed references to be provided to applicants during examination: None
Learning
Objective: DESCRIBE the major parameter changes associated with decreased RCS flow rate.
Question Source: Bank #404410
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.3, 41.5, and 41.14
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 4	Tier #	1	1
K/A Statement: Interrelation between a loss of RHR System and Service Water or Closed Cooling Water Pumps	Group #	1	1
	K/A #	APE.025.AK2.3	
Proposed Question:	Importance Rating	2.7	2.7

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

- The "A" Train of RHR is in service in the COOLDOWN mode.
- The "B" Train of RHR is in STANDBY.

The following sequence of events occurs:

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

Complete the following statement concerning the automatic action that occurs due to the high temperature condition; and the action the ARP directs the crew to take to mitigate the consequences of the event.

RHR Heat Exchanger___(1)___fails open. The crew will throttle the___(2)___.

a) (1) Total Flow Control Valve 3RHS*FV618
   (2) 3RHS*HC606 "HX A FLOW" controller in the CLOSED direction to reduce RHR flow through the RHR Heat Exchanger
b) (1) Total Flow Control Valve 3RHS*FV618
   (2) 3RHS*HC606 "HX A FLOW" controller in the OPEN direction to increase cooling to the Reactor Coolant System
c) (1) Outlet Flow Control Valve 3RHS*FV606
   (2) 3CCP*FK66A1 "RPCCW HX FLOW" controller in the CLOSED direction to prevent exceeding RPCCW System temperature limits
d) (1) Outlet Flow Control Valve 3RHS*FV606
   (2) 3CCP*FK66A1 "RPCCW HX FLOW" controller in the OPEN direction to increase cooling to the Reactor Coolant System

Proposed Answer: A

Explanation (Optional): On a loss of the RPCCW Pump, associated train RPCCW system temperature in the vicinity of the RHR Heat Exchanger will begin to increase. The RHR heat exchanger RPCCW outlet maximum operating temperature is 145°F for both normal and Safety Grade Cold Shutdown operation. "A" is correct, since when 155°F is reached on the RHR Heat Exchanger RPCCW outlet temperature with the "HX A FLOW CONT" switch in the "COOLDOWN" position, 3RHS-FCV618 (which bypasses the RHR Heat Exchanger), will fail open to minimize the heat input from RHR into the RPCCW System. Operators will be directed to adjust 3RHS-HC606, "HX A FLOW," controller closed to reduce RHR flow through the RHR heat exchanger as necessary to further minimize heat input into the RPCCW System. "B" is wrong, since operators are directed to adjust 3RHS-HC606, "HX A FLOW," controller closed to reduce RHR flow through the RHR heat exchanger to further minimize heat input into the RPCCW System. "B" is plausible, since temperature is high, and RCS decay heat removal is a concern. "C" and "D" are wrong, since 3RHS-FCV618 fails open in order to minimize the heat input from RHR into the RPCCW System, and 3RHS-FCV606 does not fail open. "C" and "D" are plausible, since FCV606 opening would increase cooling flow for the RHR System, and temperature is increasing. Also, if desired, the crew can ADJUST 3CCP*FK66A1 "RPCCW HX FLOW" controller, to increase flow as necessary (without exceeding flow limit) to reduce the RPCCW temperature at the RHR HX outlet to within design limits.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP 3353.MB2C (Rev. 0), 1-4, "Automatic Function", and step 5 LSK-27-7H (Rev. 13) and 27-7L (Rev. 1) P&ID 112A (Rev. 50)
Proposed references to be provided to applicants during examination:	None
Learning Objective:	Given a failure, partial or complete, of the residual heat removal system, DETERMINE the effects on the system and on interrelated systems.
Question Source:	Bank #404636
Question History:	Last NRC Exam 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 # 5	Tier #	1	1
K/A Statement: The reason for the conditions that	Group #	1	1
will initiate the auto opening and closing of the SWS	K/A #	APE.026.AK3.1	
isolation valves to the CCWS coolers	Importance Rating	3.2	3.5

Proposed question:

An inadvertent CDA signal occurs, causing the Service Water Supply to RPCCW Valves (3SWP*MOV50A and B) to automatically CLOSE.

Why are these valves 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 per pump. Total flow to RPCCW Heat Exchangers is 14,776 gpm, or 7,388 gpm per train, and flow to the RSS Heat Exchangers is 21,600 gpm, or 10,800 gpm per train. So with both the RPCCW Heat Exchangers and the RSS Heat Exchangers being supplied by a train of Service Water, flow from that train would be 7,388 gpm + 10,800 gpm = 18,188 gpm, which exceeds the maximum flowrate of 15,000 gpm 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 Exchanger Supply Valves (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. 34), Precaution 3.7
(Attach if not previously provided, including version/revision number.) FSAR Table 9.2-1 (Rev. 31.2)
P &ID 133A (Rev. 44)
P &ID 133B (Rev. 90)

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)…

Question Source: Bank #405265

Question History: Last NRC Exam Millstone 3 2017 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 # 6	Tier #	1	1
K/A Statement: Ability to operate or monitor	Group #	1	1
Pzr heaters, sprays, and PORVs during a Pzr	K/A #	APE.027.AA1.1	
pressure control system malfunction	Importance Rating	4.0	3.9

Proposed Question:

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

1. The Master Pressurizer Pressure Controller malfunctions with the **setpoint** step-changing from 2250 psia to 2325 psia.
2. Pzr Pressure Control System components automatically reposition.
3. The RO places the Pressurizer Master Pressure Controller in MANUAL.

Complete the following statement concerning the action the RO is required to take with the Master Pressurizer Pressure Controller in response to this failure.

The RO will push on the_____.

a) DECREASE pushbutton, closing both Pressurizer Spray Valves
b) INCREASE pushbutton, closing both Pressurizer Spray Valves
c) DECREASE pushbutton, deenergizing backup heaters "A","B", "D", and "E"
d) INCREASE pushbutton, deenergizing backup heaters "A","B", "D", and "E"

Proposed Answer: C

Explanation (Optional):

The Pressurizer Pressure Controller operates to automatically maintain a pressure reference setpoint, which is normally set to 2250 psia. When the reference setpoint fails to 2325 psia, the plant is instantly 75 psia too low. In this condition, spray valves will remain closed and heaters will energize, causing actual pressure to increase.
“A” and “B” are wrong, since the RO is required to deenergize heaters.
"A" and "B" are plausible since this would be the response if the controller setpoint had failed in the low direction, or if the controlling Pzr Pressure channel had failed in the high direction.
“C” is correct, and “D” wrong, since to stop the pressure increase, the RO is required to depress the DECREASE pushbutton to reduce the program setpoint.
"D" is plausible, since this would be true if operating the spray valve controllers, where the spray valves stroke OPEN by depressing the INCREASE pushbutton, and closed by depressing the DECREASE pushbutton.

Technical Reference(s):	OP 3353.MB4A (Rev. 7), 4-4, “Setpoint”, and “Automatic Functions”
(Attach if not previously provided,	Process Sheet 26 (Rev. J)
including version/revision number.)	Functional Sheet 11 (Rev. H)
	Functional Sheet 12 (Rev. F)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Pressurizer Pressure and Level Control System under Normal, Abnormal, and Emergency Operating conditions.

Question Source: Modified bank #404227 (Parent question attached)

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
This question is considered “modified”, since the failure has been changed from a step decrease in setpoint to a step increase in setpoint. This changes the correct answer, meeting the requirement for a modified distractor, which was the previous correct answer.

Original bank question #404227:

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.

Correct answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Question # 7	Tier #	1	1
K/A Statement: The reason for criteria for	Group #	1	1
securing/throttling ECCS during a SGTR	K/A #	EPE.038.EK3.9	
Proposed Question:	Importance Rating	4.1	4.5

A SG tube rupture has occurred in the "C" SG, and the following sequence of events occurs:

1. The crew completes cooling down and depressurizing the RCS.
2. When terminating SI, the crew inadvertently leaves the "A" SIH Pump running.

Assuming no further operator actions are taken, what effect will this error have on the plant?

a) The RCS will re-pressurize, resulting in unnecessary automatic cycling of the Pressurizer PORVs. This increases the likelihood of damaging the PORVs.
b) Tube leakage will re-initiate. The ruptured SG will eventually fill with water and the associated Atmospheric Relief Valve will lift.
c) The ruptured SG will rapidly depressurize. This will prevent the crew from being able to recover using the preferred backfill method.
d) Ruptured SG NR level will begin to decrease, causing a loss of the 8% NR level thermal layer in the ruptured SG. This will increase the radiation release.

Proposed Answer: B

Explanation (Optional):
This question includes knowledge based on OE from a simulator exam, where a crew chose to keep an SIH Pump running during a loss of all Charging while in E-3.
"A" is wrong, since a single SI pump with a shutoff head of 1550 psia will not pressurize the RCS to the PORV setpoint. "A" is plausible, since the RCS will repressurize with an SI pump running, and a concern is a relief valve lifting. Also, Millstone 3 Charging Pumps, which also provide high-head safety injection, are able to inject to above the pressure where PORVs would lift.
"B" is correct, since the SI pump has a shutoff head significantly higher than the lift setpoint of 1125 psig for the atmosphere relief valve. Unless SI is terminated, the SG will eventually overfill and the relief valve will lift, causing a radiation release.
"C" is wrong, since filling the ruptured SG will tend to raise ruptured SG pressure. "C" is plausible, since this is the basis for maintaining the thermal layer in the ruptured SG. Also, recovery using the backfill method is the preferred method of recovery.
"D" is wrong since thermal layer is controlled by the operators with Aux Feed. "D" is plausible, since E-3 series procedures caution the crew to maintain NR level (to maintain a thermal layer) in the ruptured SG

Technical Reference(s): E-3 (Rev. 26), steps 19 and 20, including Caution
(Attach if not previously provided, WOG Bkgd Doc (Rev. 2), Basis for E-3, Step 19 Caution
including version/revision number.) WOG Bkgd Doc (Rev. 2), Basis for E-3, Steps 19 and 20
Proposed references to be provided to applicants during examination: None
Learning
Objective: DISCUSS the basis of major procedure steps and/or sequence of steps in EOP 35 E-3.
Question Source: Bank #408025
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 8	Tier #	1	1
K/A Statement: Ability to explain and apply system limits and precautions during a loss of main feedwater event	Group #	1	1
	K/A #	APE.054.GEN.2.1.32	
	Importance Rating	3.8	4.0

Proposed Question:

The reactor is initially at 100% power.

The BOP operator reports main feedwater header pressure is rapidly dropping.

Based on the precautions in OP 3321, *Main Feedwater System*, what abnormal condition may be causing this event?

a) Turbine Driven Main Feedwater Pump seal supply temperature increased to 140°F.
b) Feedwater CTV "A" (3FWS*CTV41A) nitrogen pressure dropped to 200 psig.
c) A PEO inadvertently closed the HP Steam Isolation Valve to the "A" TDMFP (3TFM-V8).
d) A PEO inadvertently opened the Long Recycle Manual Isolation Valve (3FWS-V986).

Proposed Answer: D

Explanation (Optional):

"A" is wrong, since Main Feed Pump Seal supply temperature is under the 150°F limit. "A" is plausible, since cold condensate water is supplied to the Feed Pump seals, and in this distractor, seal supply water temperature is elevated, and there is a high temperature limit designed to protect the seals. If a Main Feed Pump Seal fails, high temperature feed water will be lost out of the failed seal.

"B" is wrong, since Feed CTV nitrogen is only used to reposition the valve. "B" is plausible, since nitrogen pressure is low out of its Tech Spec band, and if the valve fails closed, a loss of Main Feed Flow would occur.

"C" is wrong, since at power levels above ~25%, the Low Pressure (LP) Steam Source will begin supplying the TDMFP. At higher power levels, the LP steam source will be providing all the steam and closing the HP steam supply isolation valve will have no effect. "C" is plausible as this would cause a loss of feed event at lower power levels.

"D" is correct, since the long recycle piping is required to be isolated from the Feedwater System when a Main Feedwater Pump is running since it is not rated for feed system pressure with a Main Feedwater Pump running. If the long recycle piping bursts, feedwater pressure will drop.

Technical Reference(s): OP 3321 (Rev. 29), Precautions 3.12, 3.14, and 3.23
(Attach if not previously provided, including version/revision number.) Feedwater System Lesson Plan (FWS059C.R8CH.1 pgs 14 - 16)

Proposed references to be provided to applicants during examination: None

Learning Objective: DESCRIBE the major administrative & procedural precautions placed on the operation of the Main Feedwater AND the basis for each

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10, 43.2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 9	Tier #	1	1
K/A Statement: Ability to interpret and execute procedure steps during a station blackout	Group #	1	1
	K/A #	EPE.055.GEN.2.1.20	
Proposed Question:	Importance Rating	4.6	4.6

The crew has entered ECA-0.0, *Loss of All AC Power*, and initial conditions are as follows:

- PEOs have locally throttled open the Atmospheric Relief Valves to commence depressurizing the SGs.
- The RCS cooldown is in progress at 100°F/hr.

The following sequence of events occurs:

0100: The RO reports PZR level has shrunk to 0%.
0110: The RO reports PZR level has started rapidly increasing.
0115: The BOP reports all SG NR levels have lowered below the narrow range.
0120: The BOP reports SG pressures have reached 290 psig.

At what point was the crew first required by ECA-0.0 to stop depressurizing the SGs?

a) When the RO reported PZR level had shrunk to 0%
b) When the RO reported PZR level had started rapidly increasing
c) When the BOP reported all SG NR levels had lowered below the narrow range
d) When the BOP reported SG pressures had reached 290 psig

Proposed Answer: C

Explanation (Optional):
"A" and "B" are wrong, since the Note prior to step 20, states "PZR level may be lost and Reactor vessel upper head voiding may occur due to depressurization of SGs. Depressurization shall **NOT** be stopped to prevent these occurrences." "A" and "B" are plausible, since these are conditions that are not desirable, and are prevented from occurring during normal plant cooldowns.
"C" is correct, and "D" wrong, since the Caution prior to step 20, states, "SG NR level should be maintained GREATER THAN 8%... in at least one intact SG. If level **CANNOT** be maintained, SG depressurization should be stopped until level is restored in at least one SG."
"D" is plausible, since the goal of Step 20 is to depressurize Intact SGs to 290 psig, and this would be correct if SG levels had not dropped below the narrow range during the depressurization.

Technical Reference(s): ECA-0.0 (Rev. 35), step 20, including notes and cautions before step 20
(Attach if not previously provided, including version/revision number.) WOG Bkgd Doc (Rev. 2), for ECA-0.0, step 16
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a set of plant conditions, properly apply the Notes and Cautions of EOP 35 ECA-0.0.
Question Source: Bank # 407611
Question History: Last NRC Exam Millstone 3 2001 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10, 43.5
Comments

Examination Outline Cross-reference:	Level	RO	SRO
Question # 10	Tier #	1	1
K/A Statement: Ability to operate or monitor a control room normal ventilation supply fan during loss of offsite power	Group #	1	1
	K/A #	APE.056.AA1.18	
	Importance Rating	3.2	3.2

Proposed Question:

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

T=0:	Offsite power is lost.
T=0:	A CBI signal occurs due to the LOP.
T+1 minute:	The crew enters ES-0.1, *Reactor Trip Response*.
T+10 minutes:	The BOP Operator observes that Control Room Air Conditioning Unit 3HVC*ACU1A is running at VP1.

Complete the following statement concerning the expected suction alignment to 3HVC*ACU1A, assuming no operator actions have been taken?

Outside Air Isolation Valve 3HVC*AOV25 is_____, and Normal Supply Damper 3HVC*AOD 27A is_____.

	Outside Air Isolation Valve 3HVC*AOV25	Normal Supply Damper 3HVC*AOD 27A
a)	OPEN	CLOSED
b)	CLOSED	CLOSED
c)	OPEN	OPEN
d)	CLOSED	OPEN

Proposed Answer: A

Explanation (Optional):
This question is considered a K/A match since the Control Room Air Conditioning Units (ACUs) include the normal ventilation supply fans.
On a LOP, power is momentarily lost to the Control Room Radiation Monitors, generating a Control Building Isolation (CBI) signal. The Outside Air Isolation Valves 3HVC*AOV25/26 are normally open, supplying the normal ventilation ACUs via normal supply dampers 3HVC*AOD27A and B. AOV25 and 26 also provide a path to the Control Building Filters, which are in parallel to the normal supply dampers. The Filters are normally isolated by their own supply dampers. On a LOP, which causes a CBI signal, the outside air isolation valves receive an auto-open signal, allowing the supply fan in the running ACU to draw air into the control room envelope. This maintains control room pressure above atmospheric pressure, minimizing the potential for in-leakage of contamination into the control room. Air can be safely drawn in through this path, since the "A" Control Room Filter is automatically placed in service on a CBI to filter the outside air prior to entering the control room. As part of this automatic alignment, the filter supply dampers automatically open, and the ACU Normal Supply Dampers 3HVC*AOD 27A/B, which are in parallel with the filters, automatically close.
“A” is correct, since the Outside Air Isolation Valves remain open, and the Normal Supply Dampers automatically close.
“B” is wrong, since the Outside Air Isolation Valves remain open.
"C" and "D" are wrong, since the Normal Supply Dampers automatically close.
“B”, “C”, and “D” are plausible, since maintaining a suction path to a fan is important, and protecting the Control Room from outside contamination is important. Some dampers/valves receive an “open” signal, and others receive a “close” signal.

Technical Reference(s): OP 3314F (Rev. 35), step 4.13.1.e
(Attach if not previously provided, P&ID 151A (Rev. 33)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe operation of HVC/HVK systems under the following normal, abnormal, and emergency operation conditions.. Loss of power (LOP)…

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 11	Tier #	1	1
K/A Statement: Ability to determine and interpret	Group #	1	1
Pzr level controller, instrumentation, and heater	K/A #	APE.057.AA2.12	
indications during a loss of vital AC instrument bus	Importance Rating	3.5	3.7

Proposed Question:

The plant is operating at 100% power with the Pzr Level Control Selector Switch selected to Channel 1/2 on MB4.

The following sequence of events occurs:

1. Numerous Main Board annunciators illuminate.
2. The BOP operator reports VIAC 1 is deenergized.

Assuming the Reactor has not tripped and that NO operator actions have been taken, complete the following statement.

The Pzr Control Group Heaters are___(1)___, and the Charging Line Flow Control Valve (3CHS*FCV121) has automatically throttled in the___(2)___direction.

	(1)	(2)
a)	deenergized	open
b)	deenergized	closed
c)	energized	open
d)	energized	closed

Proposed Answer: A

Explanation (Optional):
VIAC 1 provides power to Pzr Level Control Channel 1, which fails low on loss of power.
"A" is correct, since the Pzr Heaters will deenergize on 1 of 2 channels failing low (and Letdown will have automatically isolated), and with the Controlling Channel indicating low, 3CHS*FCV121 will throttle in the open direction to attempt to restore Pzr level.
"B" is wrong, since with the Controlling Channel indicating low, 3CHS*FCV121 will throttle in the open direction to attempt to restore Pzr level. "B" is plausible, since this would be true if the Controller were responding to actual Pzr level, which is increasing due to letdown isolating, or if the Channel had failed high, or if the backup channel had failed.
"C" and "D" are wrong, since with either the controlling OR backup channel failing low, Letdown will isolate and the Pzr Heaters will deenergize to protect them from operating while uncovered. "C" and "D" are plausible, since this would be true if the instrument failed high, or if the coincidence for heaters deenergizing was 2/2, which is true of numerous interlocks, such as the VCT Low Level auto-swapover of the Charging Pump Suction path.

Technical Reference(s): AOP 3564 (Rev. 12), steps 1, 2, and 4.a and 4.b
(Attach if not previously provided, AOP 3571 (Rev. 16), Att. C, steps C.4 and C.7
including version/revision number.) Functional Sheet 11 (Rev. H)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure, partial or complete, of the Pressurizer Pressure and Level Control System, determine the effects on the system and on interrelated systems.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 12	Tier #	1	1
K/A Statement: Operational Implications of battery charger equipment and instrumentation during a loss of DC Power	Group #	1	1
	K/A #	APE.058.AK1.1	
Proposed Question:	Importance Rating	2.8	3.1

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

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

Complete the following statement concerning the potential impact this operation may have per the precautions of OP 3345C, and the required response of the PEO to this concern.

An in-rush of current may___(1)___. If this occurs, the PEO is required to___(2)___.

a) (1) damage the Charger 4 rectifier stack
   (2) open the Battery Output Breaker, and then energize the DC Bus from the Swing Charger
b) (1) damage the Charger 4 rectifier stack
   (2) close the Swing Charger AC Input Breaker, and then energize the DC Bus from the Swing Charger
c) (1) trip the Charger 4 DC Output Breaker
   (2) wait at least 5 minutes, and then cycle both the DC Output breaker and the Charger 4 Supply Breaker to off and on
d) (1) trip the Charger 4 DC Output Breaker
   (2) as promptly as possible, cycle both the DC Output breaker and the Charger 4 Supply Breaker to off and on

Proposed Answer: D

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

Technical Reference(s): AOP 3563 (Rev. 15), Att. D, step D.1.2
(Attach if not previously provided, including version/revision number.) OP 3345C (Rev. 16-10), Precaution 3.6
OP 3345C (Rev. 16-10), Section 4.16, including Note prior to step 4.16.4

Proposed references to be provided to applicants during examination: None

Learning Objective: DISCUSS the basis of major precautions, procedure steps/or sequence of steps (in AOP 3563)

Question Source: Bank #406993

Question History: Last NRC Exam Millstone 3 2013 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.8, 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 13	Tier #	1	1
K/A Statement: Ability to recognize abnormal indications which are entry level conditions for EOPs and AOPs during a loss of Service Water event	Group #	1	1
	K/A #	APE.062.GEN.2.4.4	
	Importance Rating	4.5	4.7

Proposed Question:

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

1. The "A" and "B" Service Water (SWP) Pumps trip.
2. The RO attempts to start the "C" and "D" SWP Pumps, and they immediately trip upon starting.
3. The crew enters AOP 3560, *Loss of Service Water*.

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

a) Monitor RCP Trip Criteria per the AOP 3560 Foldout Page, attempt to restore at least one SWP Pump locally, and proceed in AOP 3560.
b) Isolate Charging and Letdown, trip the Reactor, and go to E-0, *Reactor Trip or Safety Injection*.
c) Trip the Reactor, go to E-0, *Reactor Trip or Safety Injection*, and continue to monitor RCP Trip Criteria per the AOP 3560, *Loss of Service Water* Foldout Page.
d) Trip the Reactor, go to E-0, *Reactor Trip or Safety Injection,* and continue to monitor RCP Temperatures per AOP 3561, *Loss of Reactor Plant Component Cooling Water*.

Proposed Answer: B

Explanation (Optional):
"B" is correct, and "A", "C", and "D" wrong, since on a loss of all Service Water Pumps, AOP 3560 requires the crew to isolate charging and letdown to prevent a rapid VCT heatup, and then to trip the reactor and transition to E-0.
"A" is plausible, since these actions are correct for a loss of one train of SWP.
"C" is plausible, since RCP Trip Criteria are part of AOP 3560 Foldout Page items.
'D' is plausible as AOP 3560 transitions to AOP 3561 for a loss of one train of Service Water, since RPCCW has lost its Service Water cooling medium. And in AOP 3561, the foldout page monitors RCP Bearing and Seal Temperatures.

Technical Reference(s): AOP 3560 (Rev. 12), step 2.a.RNO
(Attach if not previously provided, including version/revision number.) AOP 3561 (Rev. 19) Foldout Page

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the conditions which require transition to other procedures (from AOP 3560)

Question Source: Bank #406928

Question History: Last NRC Exam Millstone 3 2001 NRC Exam

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 14	Tier #	1	1
K/A Statement: Ability to determine and interpret when to commence a plant shutdown if instrument air pressure is decreasing on a loss of instrument air	Group #	1	1
	K/A #	APE.065.AA2.5	
	Importance Rating	3.4	4.1

Proposed Question:

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

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

Assuming the rate of IAS pressure decrease has not changed throughout the event, at what time did AOP 3562 first require the crew to manually trip the Reactor?

a) 1412
b) 1414
c) 1415
d) 1417

Proposed Answer: C

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

Technical Reference(s): AOP 3562 (Rev 17), step 1
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss conditions which require transition to other procedures (from AOP 3562)
Question Source: Bank #406979
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 15	Tier #	1	1
K/A Statement: The reason for actions contained in the AOP for generator voltage and electric grid disturbances	Group #	1	1
	K/A #	APE.077.AK3.2	
	Importance Rating	3.6	3.9

Proposed Question:

The plant is at 80% power, and the following initial conditions exist:

- A load increase is in progress per OP 3204, *At Power Operations*.
- Turbine control is on the Load Limiter.

The following sequence of events occurs:

1. The grid becomes unstable.
2. Per AOP 3581, *Immediate Operator Actions*, the BOP presses and holds the "STANDBY SIGNAL MATCH" button in ON while rotating the STANDBY LOAD SET knob until the "IN STANDBY" light illuminates at the EHC Insert on MB7.
3. The crew enters AOP 3579, *Response to Turbine Runback/Loss of Turbine Load*.

What is the reason the crew was directed to place EHC "IN STANDBY"?

a) Improve system stability by placing stage pressure feedback in service
b) Minimize load swings by removing automatic speed and load control
c) Provide more effective turbine control by providing a faster response to manual turbine load adjustments
d) Stabilize the plant by stopping any load change evolutions that may have been in progress

Proposed Answer: B

Explanation (Optional):
"A" is wrong, but plausible, since Stage Pressure Feedback (SPF) is automatically inserted by the EHC system when the operator depresses the control valve test pushbutton during turbine valve testing to maintain a stable load condition while a Control Valve is stroked.
"B" is correct, since Standby Load Control Circuit is a manual system that allows turbine operation without automatic speed or load control. Selecting Standby limits load swings as grid frequency changes.
"C" is wrong, but plausible, since this is an advantage of operating on the load limiter rather than load set.
"D" is wrong, since this is accomplished by while on the load limiter by lowering the load limit pot setpoint below current turbine load. "D" is plausible, since a load increase is in progress.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
AOP 3581 (Rev. 5), Att. A, step A.1.b.RNO
AOP 3579 (Rev. 6), step 1.RNO
AOP 3579 Basis Document (Rev. 6), step 1
OP 3323A (Rev. 20), Notes and Cautions for Sections 4.6, 4.7, 4.11

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basis of major precautions, procedure steps, and/or step sequence in AOP 3579

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 16	Tier #	1	1
K/A Statement: Interrelation between a LOCA outside CTMT and heat removal systems, including RCS, ECCS, RHR, and relation between proper operation of those systems to the operation of the facility	Group #	1	1
	K/A #	W/E04.EK2.2	
	Importance rating	3.8	4.0

Proposed Question:

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

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

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

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

Proposed Answer: A

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

Technical Reference(s): ECA-1.1 (Rev 19), Step 23, and associated NOTE.
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major action categories within EOP 35 ECA-1.1

Question Source: Bank #407668

Question History: Last NRC Exam Millstone 3 2015 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 # 17	Tier #	1	1
K/A Statement: Ability to determine and interpret facility conditions and selection of appropriate procedures during a loss of emergency coolant recirculation	Group #	1	1
	K/A #	W/E11.EA2.1	
	Importance Rating	3.4	4.2

Proposed Question:

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

- The crew has completed ES-1.3, *Transfer to Cold Leg Recirculation*.
- The crew is currently in E-1, *Loss of Reactor or Secondary Coolant*.

The following sequence of events occurs:

1. The RO reports all four Containment Recirculation (RSS) Pumps show signs of cavitation.
2. Initial operator actions per the Foldout Page are NOT successful at mitigating the cavitation.

Per the Foldout Page of E-1, to which procedure is the crew required to transition?

a) ES-1.2, *Post LOCA Cooldown and Depressurization*
b) ES-1.3, *Transfer to Cold Leg Recirculation*
c) ES-1.4, *Transfer to Hot Leg Recirculation*
d) ECA-1.1, *Loss of Emergency Coolant Recirculation*

Proposed Answer: D

Explanation (Optional):
This question is considered RO level, since Millstone 3 has a RO level objective to properly apply foldout page items of E-1. Also, it requires only knowledge of the mitigation strategies of the EOPs listed in the distracters to determine the correct transition. Per the E-1 Foldout Page, if indications of sump blockage occur, the crew will stop the two RSS pumps providing only CTMT Spray. If this does not stop the cavitation on the remaining pumps, the crew will stop all Charging and all SIH Pumps, and then the "A" and "B" RSS Pumps.
"A", "B", and "C" are wrong, and "D" is correct, since with less than one Charging and one SIH Pump running, the crew is required to transition to ECA-1.1, *Loss of Emergency Coolant Recirculation*.
"A" is plausible, since ES-1.2 directs the crew to Cooldown to Cold Shutdown and place an RHR Pump in service in the Cooldown mode, which is a strategy attempted in ECA-1.1.
"B" is plausible, since ES-1.3 attempts to align at least one train of ECCS for Cold Leg Recirculation.
"C" is plausible, since the next expected transition from E-1 (if sump blockage issues were mitigated) would be to ES-1.4. This is also the correct transition if in core debris blockage were occurring.

Technical Reference(s): E-1 (Rev. 27), Foldout Page, Recirc Sump Screen Blockage Criteria
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of E-1.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.10, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 18	Tier #	1	1
K/A Statement: Interrelation between a loss of secondary heat sink and control and safety systems including instrumentation, signals, interlocks, failure modes, and automatic and manual features	Group #	1	1
	K/A #	W/E05.EK2.1	
	Importance Rating	3.7	3.9

Proposed Question:

The plant trips from 100% power, resulting in the following sequence of events:

1. The crew enters FR-H.1, *Loss of Secondary Heat Sink* from E-0, step 9.
2. The crew commences attempts to restore feed from the Motor Driven Main Feed Pump.

What actions are physically required by the crew in order to meet the interlocks to allow the crew to reset the Feedwater Isolation signal in preparation for restoring Main Feedwater flow to the Steam Generators?

a) Reset SIS only.
b) Reset P-4 only.
c) Reset SIS and pull the RPS universal logic cards.
d) Reset P-4 and pull the RPS universal logic cards.

Proposed Answer: C

Explanation (Optional):
Since the crew entered FR-H.1 from E-0, step 9, SIS has actuated, and has not been reset prior to entering FR-H.1. SIS generates a FWI signal. The SIS signal combines with the P-4 signal ("B" plausible) to lock in the FWI Signal. SIS must be reset in order to reset the FWI, and either P-4 must be reset, or the universal logic cards must be pulled to clear the lock-in feature of the FWI
"A" and "B" are wrong, since with a SIS signal, the universal logic cards must be removed to remove the seal-in feature for the FWI. "A" and "B" are plausible, since P-4 without SIS also generates a FWI signal, and if the FWI was due to P-4 without SIS, the universal logic cards would not need to be removed.
"C" is correct, and "D" wrong, since with a SIS, the SIS signal needs to be reset, and either the universal logic cards must be removed or P-4 reset to reset the FWI. "D" is plausible, since P-4 also generates a FWI signal.

Technical Reference(s): FR-H.1 (Rev. 26), step 6.b
(Attach if not previously provided, including version/revision number.) E-0 (Rev. 32), step 9
Functional Sheet 13 (Rev. K)
Proposed references to be provided to applicants during examination: None
Learning Objective: DESCRIBE the operation of the following Main Feedwater Systems Controls & Interlocks…Main Feed Containment Isolation Trip Valves (FWS*CTV41A/B/C/D)… Low Tave/P-4 Interlock, Safety Injection Signal Actuation (as related to the Main Feedwater System), Feedwater Isolation Reset Switches (MB2 & MB5)
Question Source: Bank #408172
Question History: Last NRC Exam Millstone 3 2004 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 19	Tier #	1	1
K/A Statement: Ability to operate or monitor the Bank Select Switch as it applies to a continuous rod withdrawal event	Group #	2	2
	K/A #	APE.001.AA1.1	
	Importance Rating	3.5	3.2

Proposed Question:

With the plant initially at 48% power with a plant shutdown in progress, the following initial sequence of events occurs:

1. Control Bank D rods start automatically withdrawing in the outward direction.
2. While attempting to place rods in MANUAL, the RO inadvertently places the Bank Selector Switch in the "Control Bank D" (CBD) position, and the rods stop withdrawing.
3. The crew enters AOP 3552, *Rod Control Malfunction*.
4. I&C repairs the Rod Control System.
5. While in the CBD position, the RO inserts rods from 165 steps to 150 steps.
6. Per the direction of the US, the RO places Rod Control back in Automatic.

If NO further action is taken with respect to the Rod Control system malfunction, what will be the effect on Rod Control as rods are inserted while the plant is being shutdown?

a) Control Bank C rods will begin to insert sooner than expected.
b) Control Bank C rods will begin to insert later than expected.
c) The ROD CONTROL LIMIT LO LO annunciator will illuminate at a rod height above the required value.
d) The ROD CONTROL LIMIT LO LO annunciator will illuminate at a rod height below the required value.

Proposed Answer: B

Explanation (Optional):
In "Bank Select", the Bank Overlap Unit is frozen. The Bank Overlap Circuit determines when the individual banks start to insert in MANUAL or AUTO. With the Bank Overlap Unit frozen while the crew inserted rods (while in "Bank Select"), it did not detect Bank D rod height decreasing from 165 to 150 steps,
"A" is wrong, and "B" correct, since when rods are inserted during the shutdown, the Bank Overlap Unit has Control Bank D rods 15 steps higher than they actually are. So, Control Bank C rods will start to insert 15 steps later than expected. "A" is plausible, since rods have been moving in and out, and there was a failed input to the Bank Overlap Unit.
The P/A converter feeds the RIL computer and "Bank D Full Withdrawal Limit C-11" annunciator.
"C" and "D" are wrong, since the P/A Converter still received the outward demand pulses while in "Bank Select", and the RIL setpoint, which is based on Auctioneered ΔT, also functioned correctly.
"C" and "D" are plausible, since this would have been affected if Bank Select froze the P-A converter.

Technical Reference(s): OP 3302A (Rev. 16), Precaution 3.6
(Attach if not previously provided, Functional Sheet 9 (Rev. H)
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the following Rod Control System controls and interlocks… Bank Selector Switch…
Question Source: Modified Bank #404806 (Parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.10
Comments:
Question is considered "modified" since the direction the crew moved the rods while in "Bank Select" was changed from outward to inward. This created a new correct answer.

Original Bank Question

Initial Conditions:

- The plant is at 80% power.
- Bank D Control Rods are at 186 steps.
- Preparations are being made for a load increase.

The following sequence of events occurs:

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

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

a) Control Bank C rods will begin to insert early, with Bank D rods well above their proper setpoint.
b) Control Bank C rods will begin to insert late, with Bank D rods well below their proper setpoint.
c) The calculated RIL will cause the ROD CONTROL LIMIT LO-LO annunciator to come in with Bank D rods well above the proper setpoint.
d) The calculated RIL will cause the ROD CONTROL LIMIT LO-LO annunciator to come in with Bank D rods well below the proper setpoint.

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Question # 20	Tier #	1	1
K/A Statement: Adherence to appropriate procedures and operation with the limitations in the facility's license and amendments during a SG overpressure event	Group #	2	2
	K/A #	WE13.AA2.2	
	Importance Rating	3.0	3.4

Proposed Question:

The crew has transitioned from ES-0.1, *Reactor Trip Response* to FR-H.2, *Response to Steam Generator Overpressure* due to an overpressure condition in "B" SG, and current conditions are as follows:

- "B" SG pressure is 1240 psig.
- "B" SG level is 60%.
- RCS Tave is 570°F.
- The crew was NOT successful in dumping steam from the "B" SG.

What action are the operators required to take in accordance with FR-H.2?

a) Return to procedure and step in effect.
b) Maximize Blowdown flow from "B" SG.
c) Maximize Auxiliary feed flow to the "B" SG.
d) Dump steam from the "A", "C", and "D" SGs.

Proposed Answer: D

Explanation (Optional):
The quickest way to reduce pressure in the affected SG is to dump steam from it. This has been unsuccessful.
"A" is wrong, since FR-H.2 does not have a transition step after failing to dump steam from the affected SG. "A" is plausible, since FR-H.2 would transition the crew out of the procedure if dumping steam from the affected SG was successful per step 5m and also if SG level were high, per step 3. Also, FR-H.4, *Response to Loss of Normal Steam Release Capabilities* transitions the crew back to the procedure and step in effect if unsuccessful at dumping steam from the affected SG.
"B" is wrong, since increasing Blowdown flow is not attempted in FR-H.2. "B" is plausible, since using Blowdown flow is an action in FR-H.3, *Response to Steam Generator High Level*, and it would reduce SG pressure, and the "B" SG is the affected SG.
"C" is wrong, since FR-H.2 cautions the crew not to feed the affected SG until steam release capability is restored. "C" is plausible, since feeding the SG would cool the SG, but FR-H.2 cautions against feeding the affected SG until a steam heat removal path is made available. Also, the "B" SG is the affected SG.
"D" is correct, since FR-H.2 will direct the crew to dump steam from the unaffected SGs to reduce RCS temperature, which will reduce affected SG pressure.

Technical Reference(s): FR-H.2 (Rev. 9), steps 1-7, including step 7.a.RNO
(Attach if not previously provided, including version/revision number.) FR-H.4 (Rev. 9), steps 1-3
FR-H.3 (Rev. 10), step 9
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories within EOP 35 FR-H.2.
Question Source: Bank #408258
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 21	Tier #	1	1
K/A Statement: Operational Implications of the use of steam tables during a SG tube leak	Group #	2	2
	K/A #	APE.037.AK1.1	
Proposed Question:	Importance Rating	2.9	3.3

The crew has entered AOP 3576, *Steam Generator Tube Leak*, and current conditions are as follows:

- All Control Rods: Fully inserted, except for one rod, which is stuck fully out
- RCS pressure: 900 psia and slowly decreasing
- CETC's: 505°F and slowly decreasing
- Pzr level: 20% and stable
- VCT level: 41% and stable

Based on the foldout page of AOP 3576, what action is required to be taken by the crew?

a) Shift Charging Pump suction to the RWST.
b) Adjust Charging Flow Control Valve to raise Pzr level to 50%.
c) Go to AOP 3566, *Immediate Boration*.
d) Actuate SI and go to E-0, *Reactor Trip or Safety Injection*.

Proposed Answer: D

Explanation (Optional):

"A" is wrong, since shifting Charging Pump suction to the RWST is not required unless VCT level cannot be maintained. "A" is plausible, since VCT level is in the low end of the band, and VCT level is one of the foldout page criteria.
"B" is wrong, since raising Pzr level is not required unless level is decreasing. "B" is plausible, since Pzr level is low, and it is listed as a continuous action on the foldout page.
"C" is wrong, since AOP 3566 entry is not part of the foldout page items of AOP 3576. "C" is plausible, since initiating immediate boration is required in AOP 3576, step 11 after shutdown banks have been inserted, and one rod is stuck out.
"D" is correct, since the plant is in MODE 3 (Rods are inserted), and subcooling is about 27°F. With subcooling less than 32°F, the crew is required to initiate SIS and go to E-0.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	AOP 3576 (Rev. 8), Foldout Page SI Actuation Criteria (MODE 3) AOP 3576 (Rev. 8), steps 10 and 11 Steam Tables
Proposed references to be provided to applicants during examination:	**Steam Tables**
Learning Objective:	Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of AOP 3576.
Question Source:	New
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.8, 41.10
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 22	Tier #	1	1
K/A Statement: Ability to diagnose and recognize trends in an accurate and timely manner utilizing control room reference material during a loss of condenser vacuum	Group #	2	2
	K/A #	APE.051.GEN.2.4.47	
	Importance Rating	4.2	4.2

Proposed Question:

With the plant in HOT STANDBY, a PEO is dispatched to investigate a slowly decreasing condenser vacuum.

The PEO and BOP operator report the following:

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

What is the likely cause of the decreasing vacuum?

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

Proposed Answer: B

Explanation (Optional):
"A" is wrong since one circ pump per bay is adequate in HOT STANDBY, and excessive flow out of the air ejector exists, indicating non-condensable gasses are accumulating in the main condenser. “A” is plausible, since Circ Water flow is abnormally low, and could be the problem at 100% power.
"B" is correct, since the normal gland sealing steam pressure band is 3 to 5 psig, and the AOP checks for between 3 and 5 psig. Current gland sealing steam pressure is below this, and the AOP will direct action to restore gland sealing steam pressure.
"C" is wrong, since SJAE flow is greater than normal. “C” is plausible, since a loss of air ejector steam would result in condenser vacuum decreasing.
"D" is wrong since indications of backfiring include elevated SJAE suction manifold temperature, and/or reduced or no flow out of SJAE atmospheric discharge, and per the question stem, excess flow exists.
“D” is plausible, since condenser vacuum is decreasing, and this is also a symptom of backfiring. Also, flow is abnormal through the SJAE.

Technical Reference(s): AOP 3559 (Rev. 13), steps 5 and 6
(Attach if not previously provided, including version/revision number.) OP 3329 (Rev. 12), Precaution 3.1

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Main Turbine, Gland Seal, and Gland Exhaust Systems.

Question Source: Bank #406912

Question History: Last NRC Exam Millstone 3 2002 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 23	Tier #	1	1
K/A Statement: Interrelation between	Group #	2	2
a loss of 4KV bus and associated breakers or	K/A #	Site Priority 3577.AK2.1	
supplied loads	Importance Rating	3.5	3.9

Proposed Question:

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

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

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

a) Locally shift the RCP seal return path to the top of the VCT.
b) Simultaneously isolate Charging and Letdown.
c) Transfer Non-Vital Instrument Bus 6 to the Alternate AC Source.
d) Isolate Auxiliary Steam to the Auxiliary Building.

Proposed Answer: A

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

Technical Reference(s): AOP 3577 (Rev. 5), Steps 1, 8, 9, and 15
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major action categories contained within AOP-3577
Question Source: Bank #406942
Question History: Last NRC Exam 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 # 24	Tier #	1	1
K/A Statement: The reason for manipulation of controls required to obtain desired operating results during an inadequate/degraded core cooling event	Group #	2	2
	K/A #	W/E06.EK3.3	
	Importance Rating	4.0	3.9

Proposed Question:

The crew is progressing through FR-C.1, *Response to Inadequate Core Cooling*, and current conditions are as follows:

- Core Exit Thermocouples indicate 1250°F.
- All four SG Narrow Range levels indicate 50%.
- The current step of FR-C.1 directs the crew to "Check If RCPs Should Be Started."

How will this step be implemented, and why?

a) All RCPs are started at once to supply crossover leg water to the core.
b) All RCPs are started at once to improve heat transfer to the SGs.
c) One RCP is started at a time to maximize the time cooling is provided. This temporarily cools the core by supplying it with crossover leg water.
d) One RCP is started to reduce RCS pressure. This allows low pressure safety injection from the SIL Accumulators and from the RHR pumps.

Proposed Answer: C

Explanation (Optional):
"A" and "B" are wrong, since for this step in FR-C.1, one RCP is started at a time, not all RCPs at once.
"A" and "B" are plausible, since starting all RCPs at once would initially provide the greatest cooling flow to the core.
"C" is correct, "D" wrong, since per the background document, starting an RCP (with adequate heat sink) will force two phase flow through the core, temporarily keeping it cool. The RCPs take a suction on the crossover legs; and starting the pumps one at a time extends the time this temporary core cooling method will be effective. "D" is plausible, since this is the basis for depressurizing the secondary, which has already been tried at this point in FR-C.1, and for opening Pzr PORVs, which will be tried after all RCPs have been started, or if there is inadequate heat sink. The reason heat sink is required in loops where an RCP is started is that when water enters the core, superheated steam from the core will enter the SG U-tube area, and if the tubes are not covered, rapid tube creep failure could occur due to high temperature if the tubes are not covered on the secondary side.

Technical Reference(s): FR-C.1 (Rev. 18), step 18
(Attach if not previously provided, including version/revision number.) FR-C.1 Background Doc (Rev. 2), Section 2.3, and step 18.
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 FR-C.1
Question Source: Bank #408126
Question History: Last NRC Exam Millstone 3 2011 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 25	Tier #	1	1
K/A Statement: Ability to operate or monitor components, control and safety systems including instrumentation, signals, interlocks, failure modes, auto and manual features during a CTMT flooding event	Group #	2	2
	K/A #	W/E15.EA1.1	
	Importance Rating	2.9	3.0

Proposed Question:

With the plant initially at 100% power, a LOCA occurs in CTMT, and the RCS rapidly depressurizes to CTMT pressure. Current conditions are as follows:

- The crew has entered EOP 35 FR-Z.2, *Response To CTMT Flooding*.
- Per FR-Z.2 step 1, the crew is trying to identify and isolate the sources of water to the CTMT sump.

The US directs the RO to check the Reactor Plant Component Cooling Water, Fuel Pool Cooling and Purification, and Fire Protection Water systems as potential water sources into Containment.

Which of these three paths can the RO monitor from the mainboards versus having to dispatch a PEO to check the path locally? And, assuming all systems have operated as designed, which of these paths, if any, still needs to be isolated from CTMT?

a) All CTMT isolation valves for all three systems can be verified closed at MB1. All three paths are expected to already be isolated.
b) The SFC path can only be verified locally, and the FPW CTMT outer isolation valve has a bypass valve that can only be verified locally. All three paths are expected to already be isolated.
c) All CTMT isolation valves for all three systems can be verified closed at MB1. The RPCCW System still needs to be isolated at MB1.
d) The FPW CTMT outer isolation valve has a bypass valve that can only be verified locally. The FPW bypass valve still needs to be isolated, requiring local manual operation at the CTMT penetration area.

Proposed Answer: B

Explanation (Optional):
On a large break LOCA, SIS, CIA, CDA, and CIB have actuated. The FPW system automatically isolates on a SIS/CIA, and the RPCCW System automatically isolates on a CIB.
"A" and "C" are wrong, since the FPW system has a manual bypass valve (3FPW*V666) around the CTMT isolation valve and the SFC system has manual isolation valves that can only be checked locally.
“A” is plausible, since most systems have all of their CIA valve indications on MB1.
“C” is plausible, since the RPCCW path does not isolate on a CIA..
"B" is correct and "D" wrong, since the FPW and SFC manual valves are procedurally kept closed in MODE 1. “D” is plausible, since the FPW path has a manual isolation valve, and Fire Protection Water is important to have available for use inside CTMT.

Technical Reference(s): FR-Z.2 (Rev. 6), step 1
(Attach if not previously provided, including version/revision number.) P&ID 111A (Rev. 39), 121B (Rev. 21), and 146B (Rev. 51)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the following RPS controls and interlocks… ESF Actuation Signals 1. Safety Injection… 2. CDA Actuation Signals… 3. CIA Actuation Signals 4. CIB Actuation Signals…
Question Source: Modified Bank #408245 (parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7
Comments:

This question is considered "modified", since the stem was changed with the US directing the RO to check the RPCCW System versus the Blowdown System. Distractor "C" has been modified, with the crew checking the RPCCW path, rather than the Blowdown path.

Original Bank question #408245

Current conditions:

With the plant initially at 100% power, a large break LOCA occurs in CTMT, and CDA actuates.

- The crew has entered EOP 35 FR-Z.2 *Response To CTMT Flooding*, and is trying to identify and isolate the sources of water to the CTMT sump per FR-Z.2 step 1.
- The RO has been directed to check the Steam Generator Blowdown, Fuel Pool Cooling and Purification, and Fire Protection Water systems as potential water sources into Containment.

How will the crew check these three paths, and which of these paths, if any, should still need to be isolated from CTMT?

a) The CTMT isolation valves for all three systems can be verified closed at MB1. All three paths should already be isolated.
b) The SFC path can only be verified locally, and the FPW CTMT outer isolation valve has a bypass valve that can only be verified locally. All three paths should already be isolated.
c) The CTMT isolation valves for all three systems can be verified closed at MB1. The Blowdown System still needs to be isolated.
d) The FPW path can only be verified locally. The FPW CTMT path still needs to be isolated, requiring a local manual CTMT isolation valve to be closed at the CTMT penetration area.

Correct answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Question # 26	Tier #	1	1
K/A Statement: Interrelation between a LOCA cooldown and depressurization and heat removal systems including RCS, ECCS, RHR, and relation between proper operation of those systems to the facility	Group #	2	2
	K/A #	W/E03.EK2.2	
	Importance Rating	3.7	4.0

Proposed Question:

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

- The crew is performing actions in ES-1.2, *Post LOCA Cooldown and Depressurization*.
- Both RHR pumps have been stopped
- Both Charging Pumps and both SIH Pumps are running.
- RCS pressure: 600 psia and stable.
- PZR Level: 20% and stable
- CTMT Pressure: 20 psia
- CTMT Temperature: 188°F
- CTMT Rad levels: 10 R/hr

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

Time	Event
0900:	Pressurizer level decreases to less than 16%.
0902:	Pressurizer level decreases to less than 9%.
0906:	RCS Pressure drops to less than 500 psia.
0908:	RCS Pressure drops to less than 300 psia.

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

a) 0900
b) 0902
c) 0906
d) 0908

Proposed Answer: C

Explanation (Optional):
To provide adequate ECCS flow, RCS pressure is monitored to ensure that the RHR pumps are manually restarted if pressure decreases to LESS THAN 300 psia or LESS THAN 500 psia if CTMT conditions are "Adverse".
"A" and "B" are wrong, since the restart criteria are based on RCS pressure, not Pzr level. "A" and "B" are plausible, since PZR level setpoints of 16% and 9% are both SI actuation/reinitiation setpoints in the EOP network, depending on whether the RCS could have depressurized during the accident. BUT, SI reinitiation criteria do not apply until SI has been terminated, and SIS has not been terminated at this point in ES-1.2.
"C" is correct, and "D" wrong, since adverse containment conditions exist with CTMT temperature above 180°F, where the pressure setpoint becomes 500 psia.
"D" is plausible, since if Adverse Containment conditions did not exist, the low pressure setpoint is 300 psia, and CTMT radiation is well below the Adverse CTMT setpoint.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	E-1 (Rev. 27), Caution prior to step 8 ES-0.1 (Rev. 29), Foldout Page SI actuation criteria ES-1.2 (Rev. 22), Foldout page SI re-initiation criteria OP 3272 (Rev. 10), Section 1.6, page 16

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, properly apply the notes, cautions, and foldout page items of E-1.

Question Source: Bank #407943

Question History: Last NRC Exam Millstone 3 2004 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 # 27	Tier #	1	1
K/A Statement: Ability to determine and interpret	Group #	2	2
adherence to appropriate procedures and operation	K/A #	W/E08.EA2.2	
within the limitation in the facility's license	Importance Rating	3.5	4.1

During a pressurized thermal shock event

Proposed Question:

A steamline break has occurred inside CTMT, and initial conditions are as follows:

- The crew is performing actions in FR-P.1, *Response to Imminent Pressurized Thermal Shock Condition*.
- Narrow range levels for the intact SGs are 25%.
- RCS temperature is stable.
- RCS pressure is stable.
- Only the control group of Pressurizer heaters is energized.
- The Pressurizer is saturated.

The crew has just determined that a 1-hour soak is required.

Per FR-P.1, which of the following evolutions would the crew be allowed to perform in the next hour?

a) Energizing additional Pressurizer Heaters.
b) Using Auxiliary Spray.
c) Increasing AFW flow to raise SG levels to 50%.
d) Initiating a RCS cooldown at a rate < 50°F per hour.

Proposed Answer: B

Explanation (Optional):
"A" is wrong since RCS pressure cannot be raised during a SOAK.
"A" is plausible, since it does not lower temperature, and all distractors affect either RCS temperature or pressure.
"B" is correct, since soak requirements do not prohibit lowering RCS pressure.
"C" is wrong, since RCS temperature cannot be lowered.
"C" is plausible, since there is no restriction on SG levels, and all distractors affect either RCS temperature or pressure.
"D" is wrong, since RCS temperature cannot be lowered.
"D" is plausible, since this cooldown rate is reduced from the normal Tech Spec cooldown rate of 80°F/hr, and all distractors affect either RCS temperature or pressure.

Technical Reference(s): FR-P.1 (Rev. 16), step 23
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss the basis of major procedure steps and/or sequence of steps in EOP 35 FR-P.1

Question Source: Bank #408225

Question History: Last NRC Exam Millstone 3 2002 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 28	Tier #	2	2
K/A Statement: Ability to monitor automatic operation of the RCP's, including seal injection flow	Group #	1	1
	K/A #	003.A3.1	
Proposed Question:	Importance Rating	3.3	3.2

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

- An inadvertent Containment Isolation Actuation Phase A (CIA) Signal is received.
- The RO monitors MB3 to confirm the plant has automatically responded as expected to the CIA.

Complete the following statement concerning the expected effect the CIA signal had on RCP seal supply and return flows, assuming the reactor has not tripped, and no operator actions have been taken.

Seal Supply flow is being supplied from__(1)__, and Seal Return flow is flowing to the__(2)__.

	(1)	(2)
a)	one Charging Pump	PRT
b)	one Charging Pump	VCT
c)	the RCS	PRT
d)	the RCS	VCT

Proposed Answer: A

Explanation (Optional):
During normal operation, RCP seals are being supplied from the running Charging Pump, and the Seal return path is to the VCT.
"A" is correct, since on a CIA, the Seal Return Line Containment Isolation Valves 3CHS*MV8100 and MV8112 automatically CLOSE, while the seal supply path is unaffected. And when the seal return path isolates, a relief valve (3CHS*RV8121) opens on high pressure, directing Seal Return flow to the PRT.
"B" and "D" are wrong, since the seal return path to the VCT has isolated. "B" and "D" are plausible, since the VCT is the normal seal return path.
"C" is wrong, since the seal supply path is unaffected by the CIA. "C" is plausible, since if the seal supply path isolated, the backup seal supply path is from the RCS, which would be cooled by the RCP thermal barrier.

Technical Reference(s): P&ID 104A (Rev. 54)
(Attach if not previously provided, including version/revision number.) P&ID 103A (Rev. 29)
P&ID 102F (Rev. 17)
Proposed references to be provided to applicants during examination: None
Learning Objective: For the below listed plant events, partial or complete, describe the effects on the Chemical and Volume Control System and its interrelated systems… CTMT Isolation Signal Phase A…
Question Source: Bank #404387
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 29	Tier #	2	2
K/A Statement: Ability to monitor automatic operation	Group #	1	1
of the CVCS System, including ion exchange bypass	K/A #	004.A3.3	
Proposed Question:	Importance Rating	2.9	2.9

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

1. Letdown temperature downstream of the Letdown Heat Exchanger increases to 135°F.
2. The LETDOWN HX OUT TEMP HI annunciator is received on MB3A.
3. The RO reports Letdown Heat Exchanger temperature controller 3CHS*TK130 has automatically opened RPCCW Letdown Heat Exchanger Return Valve 3CCP*TV172, and letdown temperature is returning to normal.
4. The LETDOWN HX OUT TEMP HI annunciator clears.

The RO is continuing to monitor the Letdown path to verify proper automatic system response.

Complete the following statement, assuming Letdown temperature has returned to normal and no operator actions have been taken.

Letdown temperature can be monitored from___(1)___at MB3, and the RO will observe Letdown Demin Supply Valve 3CHS*TCV129 routing letdown to___(2)___the demineralizers.

	(1)	(2)
a)	3CHS*TE130 only	flow through
b)	3CHS*TE130 only	bypass
c)	3CHS*TE130 and 3CHS*TE129	flow through
d)	3CHS*TE130 and 3CHS*TE129	bypass

Proposed Answer: B

Explanation (Optional): “A” and “C” are wrong, since the demins are bypassed at 134ºF as sensed by 3CHS*TE129, and 3CHS*TCV129 does not automatically reposition back to the Demineralizer position after temperature drops below 134°F. “A” and “C” are plausible, since temperature exceeded 134°F by only one degree, and also temperature has returned to normal, and control valves in some systems automatically reposition when a parameter returns to normal (e. g. Service Air to Inst Air cross-tie valve). "B" is correct, and “D” wrong, since 3CHS*TE130 controls Letdown Heat Exchanger cooling and provides temperature indication on MB3, while 3CHS*TE129 sends the “divert to VCT” signal to 3CHS*TCV129, but does not supply a temperature indication on MB3. “D” is plausible, since TE129 does detect Letdown Temperature, and many parameters on the Main Boards have redundant indications available to the operators.

Technical Reference(s): OP 3353.MB3A (Rev. 4), 5-5
(Attach if not previously provided, including version/revision number.) P&ID 104C (Rev. 32)
P&ID 104A (Rev. 54)

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 30	Tier #	2	2
K/A Statement: Ability to manually operate or monitor CVCS letdown isolation and flow control valves from the control room	Group #	1	1
	K/A #	004.A4.6	
	Importance Rating	3.6	3.1

Proposed Question:

The crew has just completed cooling down the plant per OP 3208, *Plant Cooldown*, and current conditions are as follows:

1. The plant is in MODE 5.
2. The Pzr is solid.
3. RCS pressure is being maintained by Letdown Pressure Control Valve 3CHS*PCV131.

What is the current positions of the RHR System Letdown Control Valve (3CHS-HCV128) and the Letdown Orifice Isolation Valves (3CHS*AV8149A, B, and C)?

	3CHS-HCV128	3CHS*AV8149A, B, and C
a)	Fully open	All three open
b)	Fully open	One open
c)	Throttled	All three open
d)	Throttled	One open

Proposed Answer: A

Explanation (Optional):
"A" is correct, since during the cooldown, operators slowly increased the output of 3CHS-HCV128 to 100% (full open). Also, during the cooldown, operators were directed to open additional letdown orifices as necessary to maintain maximum letdown, and letdown flow decreases significantly as RCS pressure is reduced during the cooldown. When placing the Pzr in a solid condition, the crew verifies the position of these valves.
"B" is wrong, since during the cooldown, operators were directed to open additional letdown orifices as necessary to maintain maximum letdown, and letdown flow decreases significantly as RCS pressure is reduced during the cooldown. "B" is plausible, since the normal position of these valves is one open.
"C" and "D" are wrong, since during the cooldown, operators slowly increased the output of 3CHS-HCV128 to 100% (full open). "C" and "D" are plausible, since the normal position of this valve is closed, and during the cooldown, it is in the throttled position, while slowly being opened.

Technical Reference(s): OP 3208 (Rev. 31), steps 4.2.12.c, 4.3.25, and 4.4.4.b
(Attach if not previously provided, including version/revision number.) P&ID 104A (Rev. 54)

Proposed references to be provided to applicants during examination: None

Learning Objective: Restore Normal Charging and Letdown at Normal Operating Pressure

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 31	Tier #	2	2
K/A Statement: Effect of a loss or malfunction of the RHR heat exchanger on the system	Group #	1	1
	K/A #	005.K6.3	
Proposed Question:	Importance Rating	2.5	2.6

The plant is being cooled down in preparation for refueling, and current conditions are as follows:

- RCS temperature: 175°F
- RCS pressure: 350 psia
- The Pressurizer: Solid
- Charging Flow Control Valve: In "MANUAL"
- RHR System: "A" Train in service in the "Cooldown" mode
- "RHR HDR FLOW" 3RHS-FK618: In "AUTO"

A tube in the "A" RHR heat exchanger rapidly fails, causing a 150-gpm tube leak.

Complete the following statement concerning the effect of the tube leak, assuming no operator action is taken.

Actual RHR Pump flow will begin to___(1)___and RCS temperature will begin to___(2)___.

	(1)	(2)
a)	decrease	increase
b)	decrease	decrease
c)	increase	increase
d)	increase	decrease

Proposed Answer: C

Explanation (Optional): RHR pressure is higher than CCP pressure; so 150 gpm is flowing out of RHR into the RPCCW System downstream of the RHR Pump.
"A" and "B" are wrong, since with the tube leak, 150 gpm of flow that was sensed by RHR flow detector 3RHS*FT618 is now being lost into the RPCCW System. FT618 will detect less flow and send a signal to 3RHS*FCV618 to throttle open to maintain 4000 gpm total flow, increasing actual RHR Pump flow. "A" and "B" are plausible, since there is a leak between the RHR and RPCCW Systems, and the RHR total flow controller is responding to the transient. Depending on system pressures and the flow detector location, either direction of flow change is plausible.
"C" is correct, since RHR pump flow has increased, and with FCV618 opening, more flow bypasses the RHR Heat Exchanger, so less cooling of RHR flow is occurring, so RCS temperature begins to increase.
"D" is wrong, since less cooling of RHR flow is occurring, so the RCS begins to heat up. "D" is plausible, since total flow through the RHR pump has increased.

Technical Reference(s): OP 3208 (Rev. 31), steps 4.3.9, 4.3.13, and 4.3.14
(Attach if not previously provided, including version/revision number.) OP 3310A (Rev. 19), section 4.5
P&ID 112A (Rev. 50)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a failure, partial or complete, of the residual heat removal system, DETERMINE the effects on the system and on interrelated systems.

Question Source: Bank #404648

Question History: Last NRC Exam Millstone 3 2011 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 # 32	Tier #	2	2
K/A Statement: Predict/monitor changes in parameters due to operating ECCS controls including reactor vessel level	Group #	1	1
	K/A #	006.A1.14	
	Importance Rating	3.6	3.9

Proposed Question:

A small break LOCA is in progress, with current conditions as follows:

1900: The STA reports RVLMS plenum level has decreased to 19%.
1902: The crew enters FR-C.2, *Response to Degraded Core Cooling*.
1915: The crew holds a brief, and current conditions are reported as follows:

- PZR level: 0%
- Reactor head and plenum levels: 0%
- CETCs: 720°F and increasing
- RCS pressure: 1200 psia and stable
- Charging Pumps: None running
- SIH Pumps: None running

A PEO reports the "C" Charging Pump is ready to start.

Complete the following statements concerning how effective starting the Charging Pump will be at recovering Reactor Vessel level in accordance with the WOG Background document; and based on current conditions, evaluate how reactor vessel level will respond to this action.

This is the__(1)__effective core cooling method attempted in FR-C.2. Reactor Vessel level will__(2)__.

a) (1) least
   (2) not recover until the loop seal in the RCS crossover legs clear. After this, RCS pressure will decrease since steam from the vessel now has an exit path through the cold leg break
b) (1) least
   (2) slowly recover over the course of the next hour due to the limited injection capability of one Charging Pump and the severe lack of inventory in the RCS
c) (1) most
   (2) rapidly recover as high head injection flow raises RCS pressure above SG pressures, restoring heat removal from the RCS into the Steam Generators
d) (1) most
   (2) rapidly recover as high head injection forces subcooled water into the core. This rapidly cools and depressurizes the RCS, resulting in SIL Accumulator and RHR Pump injection

Proposed Answer: D

Explanation (Optional): Entry conditions are initially met for FR-C.2 based on RVLMS level. Also, temperature at 720° indicates superheated conditions, showing the actual fuel is being uncovered.
"A" and "B" are wrong since the first major action category in FR-C.1 and C.2 is to restore high-head safety injection, as it is the most effective method at restoring core cooling.
"A" is plausible, since this describes the cooling mechanism provided by venting the RCS through the PORVs, which is the least effective cooling method attempted in FR-C.1.
"B" is plausible, since this describes the basis for RCS pressure stabilizing at 1200 psia during the event as the vessel acts like a Pressurizer, which is at saturation conditions with SGs removing RCS heat. This is also related to system response to a small break LOCA with high head injection available.
"C" is wrong, since the RCS will rapidly depressurize when high head injection commences.
"C" is plausible, since if this were a small break LOCA with ECCS available and no void in the reactor vessel head, increasing injection flow above break flow would slowly raise RCS pressure. Also, this is a misapplication of the basis for the highly effective cooling method of dumping steam from the secondary, which increases heat removal from the RCS to the secondary plant.
"D" is correct, since restoring high head injection rapidly causes the injection of subcooled water into the bottom of the core, and it can be determined that the RCS is highly voided based on the presence of superheat in the core (The Pzr, RCS hot legs, vessel head, and vessel plenum are empty), the resultant core cooling causes rapid depressurization of the highly voided RCS to the point where low head injection begins. If this step is successful, the crew will transition out of FR-C.2 at step 6.

Technical Reference(s):	FR-C.2 (Rev. 14), steps 3 and 6
(Attach if not previously provided,	WOG Bkgd (Rev. 2) for FR-C.1, Section 2.1
including version/revision number.)	WOG Bkgd (Rev. 2) for FR-C.2, Section 3.1 and step 2
	Westinghouse Core Cooling MITCORE Text, page 2-23 and Figures
	Steam Tables

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

Learning Objective: APPRAISE each Operator-initiated recovery technique in its ability to restore the Core Cooling Critical Safety function.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5, 41.10, 41.14

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 33	Tier #	2	2
K/A Statement: Predict/monitor changes in parameters	Group #	1	1
due to operating PRT controls including monitoring	K/A #	007.A1.3	
PRT temperature	Importance Rating	2.6	2.7

Proposed Question:

With the plant at 100% power, the PZR REL TK TEMP HI annunciator illuminates on MB4A, resulting in the following sequence of events:

1. The US directs the RO to perform OP 3301A, *Pressurizer Relief Tank and Reactor Vessel Flange Leakoff Operations*, Section 4.2, "Restoring PRT to Normal Temperature".
2. The RO opens the Primary Grade Water Ctmt Isolation Valves (3PGS*CV8028 and CV8046) at MB1.
3. The RO opens 3PGS-AV8030 ("PRI WTR") at MB4.

Complete the following statement concerning the Main Board location(s) from where the RO can monitor PRT temperature; and the PRT parameter(s) upon which OP 3301A will direct the RO to close 3PGS-AV8030.

The RO can monitor PRT temperature from___(1)___, and the RO will close 3PGS-AV8030 based on ___(2)___.

	(1)	(2)
a)	MB4 only	PRT High Temperature only
b)	MB4 only	either PRT High Temperature or PRT High Level
c)	MB1 or MB4	PRT High Temperature only
d)	MB1 or MB4	either PRT High Temperature or PRT High Level

Proposed Answer: B

Explanation (Optional): With a PRT high temperature condition, the crew will open the PGS supply valve to the PRT, filling it with cold PGS water. This will raise PRT level and lower PRT temperature.
"A" is wrong, since the fill valve is required to be closed on either PRT high level or PRT low temperature. "A" is plausible, since the parameter the crew is attempting to lower is PRT temperature, the ARP direction is to "LOWER PRT temperature", and there is a PRT vent valve and drain valve that could be used to feed and bleed the tank until temperature is restored.
"B" is correct, since PRT temperature is monitored on MB4 only, and the crew will close the fill valve based on either PRT temperature lowered back to normal or based on PRT high level.
"C" and "D" are wrong, since PRT temperature cannot be monitored from Main Board 1. "C" and "D" are plausible, since PGS controls and Primary and Containment Drains Transfer Tanks (PDTT and CDTT) controls and level indications are located at Main Board 1.

Technical Reference(s): OP 3353.MB4A (Rev. 6), 2-2
(Attach if not previously provided, including version/revision number.) OP 3301A (Rev. 10), steps 4.2.1 and 4.2.2
P&IDs 102F (Rev. 17) and 119A (Rev. 34)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the Pressurizer Relief Tank System operation… under the following… Restoring from a high Pressurizer Relief Tank Temperature condition

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.3, 41.7, 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 34	Tier #	2	2
K/A Statement: Effect of loss or malfunction of the PRT on CTMT	Group #	1	1
	K/A #	007.K3.1	
Proposed Question:	Importance Rating	3.3	3.6

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

- The "A" Pzr PORV (3RCS*PCV455A) is leaking by and will not reseat.
- "A" PORV Block valve 3RCS*MV8000A will NOT close.

Assuming Safety Injection does not actuate, and no further operator actions are taken, to where will the excess water entering the PRT eventually be routed?

a) When the PRT overfills, the water will automatically relieve to the Primary Drains Transfer Tank (PDTT). When the PDTT overfills, it will relieve to Radioactive Liquid Waste.
b) When the PRT overfills, the water will automatically relieve to the Containment Drains Transfer Tank (CDTT). When the CDTT overfills, it will relieve to Boron Recovery.
c) When the PRT Rupture Disk fails, the water will be released to the Containment atmosphere. It will then collect in the Unidentified Leakage Sump, and from there, be pumped to the Identified Leakage Sump, which is then pumped to Radioactive Liquid Waste.
d) When the PRT Rupture Disk fails, the water will be released to the Containment atmosphere. It will then collect in the Identified Leakage Sump, and from there, be pumped to the Unidentified Leakage Sump, which is then pumped to Boron Recovery.

Proposed Answer: C

Explanation (Optional):
"A" and "B" are wrong, since the PRT does not relieve to the PDTT or CDTT. "A" and "B" are plausible, since the CDTT and PDTT receive reactor plant gaseous drains from numerous sources, and their discharge can be routed to either radioactive liquid waste or boron recovery. Also, the PRT does have a vent valve, but it automatically closes on a high pressure condition.
"C" is correct since the PRT is protected from overpressure by a rupture disk, which will rupture and release its contents to the containment atmosphere as it flashes to steam. As the steam condenses, it will collect on the CTMT walls and floor, draining into the unidentified leakage sump. Water in the unidentified leakage sump is pumped to the identified leakage sump, which is pumped to radioactive liquid waste.
"D" is wrong, since the water will initially be collected in the unidentified leakage sump. "D" is plausible, since numerous Ctmt drain paths lead directly to the identified leakage sump, and numerous borated water sources are routed to Boron Recovery.

Technical Reference(s): P&ID 102F (Rev. 17), including Note 3
(Attach if not previously provided, including version/revision number.) P&ID 106C (Rev. 46)
P&ID 107A (Rev. 28)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the Pressurizer Relief Tank System operation, or operations required, under the following normal, abnormal, or emergency operating conditions or procedures… Pressurizer Safety Valve OR Power Operated Relief Valve discharge…

Question Source: Bank #402809
Question History: Last NRC Exam Millstone 3 2011 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 35	Tier #	2	2
K/A Statement: CCW design feature or interlock which provides for the standby feature of the CCW pumps	Group #	1	1
	K/A #	008.K4.09	
Proposed Question:	Importance Rating	2.7	2.9

The plant is at 100% power, with the "C" RPCCW pump breaker in its procedurally defined "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, since 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) , since the "C" RPCCW Pump breaker is normally aligned to the "B" Train.

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.
"D" is correct, and "A" wrong, since the crew needs to move the breaker to the "A" train cubicle
"B" is wrong, since the "C" swing pump has one breaker that can be racked up in either train.
"C" is wrong, since the "C" swing pump has one breaker that is shared between the two trains.
"A", "B", and "C" are plausible, since two breaker cubicles exist, and for the Charging Pumps, the swing pump does not have its own dedicated breaker. It utilizes the associated train's ("A" or "B" Pump) Breaker.

Technical Reference(s): AOP 3561 (Rev 19), Att. A, step A.6
(Attach if not previously provided, including version/revision number.) OP 3330A (Rev 27), Section 1.2
OP 3330A (Rev 27), steps 4.9.11, 4.9.12, 4.9.23, and 4.9.24
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…
Question Source: Bank #402449
Question History: Last NRC Exam Millstone 3 2015 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 # 36	Tier #	2	2
K/A Statement: Effect of a loss or malfunction of Pzr sprays and heaters on the Pzr pressure control system	Group #	1	1
	K/A #	010.K6.3	
Proposed Question:	Importance Rating	3.2	3.6

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

Pzr control heater group 3RCS-H1C fails to the fully energized condition.

Assuming no operator action is taken, complete the following statement as to how Pzr Pressure Control System will respond to this failure.

RCS pressure will increase to the spray valve setpoint of __(1)__, after which the Pzr Spray Valves will__(2)__.

a) (1) 2275 psia
   (2) throttle open to maintain Pzr pressure at 2275 psia
b) (1) 2325 psia
   (2) throttle open to maintain Pzr pressure at 2325 psia
c) (1) 2275 psia
   (2) throttle open to stabilize pressure, and continue to slowly throttle open to lower pressure back to 2250 psia
d) (1) 2325 psia
   (2) throttle open to stabilize pressure, and continue to slowly throttle open to lower pressure back to 2250 psia

Proposed Answer: C

Explanation (Optional):
"B" and “D” are wrong, since the spray valves start to throttle open at 2275 psia, and receive a full open signal at 2325 psia. “A” and “C” are plausible, since the spray valve full-open setpoint is 2325 psia, which is below the PORV open setpoint of 2350 psia.
“A” is wrong, and “C” is correct, since the Pzr Pressure controller is a PI controller, so the longer the error exists, the more the output signal will continue to throttle open the spray valves to lower pressure, until pressure is restored to 2250 psia. "A" is plausible, since these pressures relate to spray valve operations, and this is how a “P” controller would respond. Also, Pzr PORVs cycle to maintain Pzr pressure at the PORV pressure setpoint.

Technical Reference(s): Training Lesson Plan PPL010C (Rev. 4), Figure 3
(Attach if not previously provided, including version/revision number.) Functional Sheet #11 (Rev. H)
Process Block Sheet 26 (Revision not listed)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure, partial or complete, of the Pressurizer Pressure and Level Control System, determine the effects on the system and on interrelated systems.
Question Source: Bank #404249
Question History: Last NRC Exam Millstone 3 2013 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 # 37	Tier #	2	2
K/A Statement: Effect of a loss or malfunction of the pressure detection system on the Pzr pressure control system	Group #	1	1
	K/A #	010.K6.1	
	Importance Rating	2.7	3.1

Proposed Question:

Reactor power is initially stable in the Intermediate Range at $1x10^{-8}$ amps.

Controlling Pressurizer Pressure Channel 3RCS*PT455 fails high.

Assuming no operator action is taken, how will the plant respond to the instrument failure?

a) Pzr backup heaters will energize at 2235 psia and stabilize Pzr pressure
b) The failed open PZR PORV will auto-close at 2000 psia and stabilize Pzr pressure
c) A reactor trip will occur at 1900 psia due to low Pzr pressure
d) SI actuation will occur at 1892 psia due to low Pzr pressure

Proposed Answer: D

Explanation (Optional):
When the controlling Pzr pressure channel fails high, the control system will respond to lower pressure by turning off Pzr heaters and opening the Pzr spray valves.
"A" is wrong, since backup heaters are driven by the failed controlling pressure channel.
“A” is plausible, since there are four channels, with a backup channel selected that has not failed, and the PORVs operate off of a2/4 coincidence.
B" is wrong, since PZR PORVs cycle off of 2/4 coincidence Pressurizer pressure channels, not the controlling channel, so the PORV did not open. Also, the spray valve remains open and the heaters are deenergized.
“B” is plausible, since the heaters and spray valves are driven off of the controlling pressure channel. Also, P-11 will auto-close a failed open PORV at 2000 psia.
“C” is wrong, since the low Pressurizer Pressure Reactor Trip is automatically blocked below P-7 (Reactor or Turbine <10% power).
“C” is plausible, since above 10% power, the Lo Pzr pressure trip is armed to actuate at 1900 psia.
"D" is correct, since the low Pressurizer pressure SI is active until manually blocked when Pzr pressure is reduced below P-11 (2000 psia).

Technical Reference(s): E-0 (Rev. 32), Attachment A, page 1
(Attach if not previously provided, including version/revision number.)
Functional Sheet 6 (Rev. J)
Functional Sheet 11 (Rev. H)
Functional Sheet 18 (Rev. D)
Functional Sheet 19 (Rev. D)

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #404936
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 38	Tier #	2	2
K/A Statement: Ability to manually operate or monitor bistable, trips, reset and test switches from the control room	Group #	1	1
	K/A #	012.A4.4	
	Importance Rating	3.3	3.3

Proposed Question:

Initial conditions are as follows:

- The plant is at 40% power.
- I&C is performing Reactor Trip Breaker testing.
- Both Reactor Trip Breakers are currently CLOSED.
- Both Reactor Trip Bypass Breakers are currently OPEN.

The following sequence of events occurs:

1. A single RPS Protection Set I "RCS Loop 'C' Low Flow" Reactor Trip Bistable light illuminates.
2. The crew enters AOP 3571, *Instrument Failure Response*.
3. The extra operator reports that the red GENERAL WARNING lamp on 3RPS*RAKLOGB is lit.

Which of the following events or actions would result in an immediate automatic reactor trip?

a) Placing the Train "A" Multiplexer Test Switch in "Inhibit".
b) Receiving a RPS Protection Set IV "RCS Loop 'C' Low Flow" Bistable.
c) Losing one Train "B" Solid State Protection 48 V power supply.
d) Closing the Train "B" Reactor Trip Bypass Breaker.

Proposed Answer: A

Explanation (Optional):
An automatic reactor trip occurs when both Reactor Trip Bypass Breakers are closed at the same time.
An automatic reactor trip also occurs when there is a “General Warning” on both trains of SSPS.
A “General Warning” is caused by any of the following:

- Loss of either 48V power supply.
- Loss of either 15V power supply.
- Bypass breaker closed.
- Test switch out of normal.
    - Input Error Inhibit Switch in Inhibit (logic cab).
    - Multiplexer test switch in Inhibit.
    - Logic “A” function switch out of Off (logic cab).
    - Permissive inhibit switch out of Off (logic cab).
    - Memory test switch out of Off (logic cab).
    - Mode Selector Switch in Test (relay test panel).
    - Any loose or removed circuit board cards.
    - Master Relay Selector Switch not in "OFF" (OP 3353.MB4C, 1-3A)

"A" is correct, since taking the Multiplexer Test Switch to “Inhibit” will cause a GENERAL WARNING on SSPS Train A, and with a GENERAL WARNING already on train B, a reactor trip will occur due to GENERAL WARNING on both trains of SSPS.
“B” is wrong, since the RCS Loop Low Flow trip in one of four RCS loops is blocked below P-8 (50% power). “B” is plausible, since the coincidence for a RCS loop low flow trip is 2 of 4 Bistables lit for a single RCS loop, and there is.
“C” is wrong, since the 48 Volt Power Supply has two inputs, so RPS still has power, and the GENERALWARNING received due to the loss of one of the two power supplies is on Train B, which is the same train as the GENERAL WARNING that already exists.
“C” is plausible, since losing a 48 Volt Power Supply causes a GENERAL WARNING alarm. Also, if a VIAC deenergizes, all Bistables powered from that Bistable fail to the tripped condition.
“D” is wrong, since only one of the two Bypass Breakers would be closed, and the GENERAL WARNING that is received when closing this breaker is on the same train as the existing GENERAL WARNING alarm.
“D” is plausible, since reactor trip breaker testing is in progress, and a closed bypass breaker generates a GENERAL WARNING alarm. It also feeds into the two Bypass Breakers closed reactor trip.

Technical Reference(s): OP3353.MB4C (Rev. 16), 1-3B
(Attach if not previously provided, including version/revision number.) Functional Sheet 2 (Rev. N)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following RPS controls and interlocks… Multiplexer Test Switch… Reactor Trip and Bypass Breakers… General Warning…

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7 and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 39	Tier #	2	2
K/A Statement: Operational implications of	Group #	1	1
ESFAS safety system logic and reliability	K/A #	013.K5.2	
Proposed Question:	Importance Rating	2.9	3.3

Complete the following statement as to why only 3 channels of Containment Pressure are required to input into the Containment High Pressure Safety Injection signal coincidence.

The Containment pressure transmitters_____.

a) do NOT input to any control functions
b) are Post Accident Monitor (PAM) qualified
c) are deenergize-to-actuate
d) fail in the HIGH direction

Proposed Answer: A

Explanation (Optional): Coincidence logic determines whether protective action is required. Plant conditions, as sensed by redundant channels, must be in agreement within a solid state protection system (SSPS) train to produce a safeguard signal. In most cases, a single-channel instrument failure will not cause a protective action since the trip logic requires at least two channels in the tripped condition. While preventing unnecessary plant shutdowns, this also permits on-line testing of a single channel without initiation of protective action. A key factor in determining the coincidence required, such as two-of-three (2/3) or two-of-four (2/4) coincidence, is whether the possibility exists for interaction between the protection and control uses of the specific type of instrumentation. If a channel supplies both protection and control system signals, it is assumed that (1) the failure of that instrument causes a plant transient which requires protective action, and (2) the failure prevents the channel from initiating the protective action. In addition, it is assumed that a second redundant channel fails to supply a trip signal. This situation requires a two-of-four logic to ensure that protective action occurs when a control system is being supplied by the instruments. A two-of-three logic is allowed if either of the following is true:

- No control action is derived from the protection channel, or
- The trip is a back-up trip to another form of protection (e. g. The Pzr High Level Reactor Trip backs up the Pzr High Pressure Reactor Trip, so the Pzr High Level Trip has a 2/3 coincidence.

"A" is correct, since no control functions are derived from the Containment Pressure Transmitters.
"B" is wrong, since PAM is not considered when determining coincidence. "B" is plausible, since PAM monitors are more reliable in accident conditions.
"C" is wrong, since energize or deenergize to actuate is not considered when determining coincidence.
"C" is plausible, since deenergize to actuate is conservative, the SI signal is deenergize to actuate, and the CDA signal is energize to actuate.
"D" is wrong, since failing high or low is not considered when determining coincidence.
"D" is plausible, since failing to the tripped condition is conservative.

Technical Reference(s): NNI016C Training Text (Rev. 2), Section 6, pages 13 and 14
(Attach if not previously provided, Functional Sheet 8 (Rev. K)
including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following… controls and interlocks… Safety Injection… Ctmt Hi-1…

Question Source: Bank #404888
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 40	Tier #	2	2
K/A Statement: Ability to manually operate or monitor CTMT readings of temperature, pressure, and humidity system from the control room	Group #	1	1
	K/A #	022.A4.5	
	Importance Rating	3.8	3.8

Proposed Question:

The RO is monitoring Ctmt parameters while taking rounds.

Complete the following statement concerning the Ctmt parameters being monitored.

As part of the RO's rounds, Ctmt ___(1)___ is/are verified to be within Tech Spec limits, and Ctmt humidity is displayed on 3LMS-MI22 as ___(2)___ on MB2.

	(1)	(2)
a)	pressure only	percent humidity (%)
b)	pressure only	dew point temperature (°F)
c)	pressure and temperature	percent humidity (%)
d)	pressure and temperature	dew point temperature (°F)

Proposed Answer: D

Explanation (Optional):
"A" and "B" are wrong, since Ctmt pressure and Ctmt temperature are Tech Spec required verifications.
"A" and "B" are plausible, since Ctmt pressure inputs to ESF actuation signals, and Ctmt temperature does not.
"C" is wrong, and "D" correct, since the Ctmt humidity signal is converted to a dew point indication on Main Board 2 that reads out in °F.
"C" is plausible, since it is common for humidity detectors to read out in % humidity.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
SP3670.1-001 (Rev. 41), pages 12-14
SP3670.1-001 (Rev. 41), pages 57-58
Tech Spec LCO 3.6.1.4 (Amendment 258)
Tech Spec LCO 3.6.1.5 (Amendment 258)
P&ID 154A (Rev. 26)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following Containment… System… indicators… Atmospheric temperature monitors… pressure monitors… humidity monitors…

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 41	Tier #	2	2
K/A Statement: Ability to monitor automatic operation of the CTMT spray system , including pump starts and correct MOV positioning	Group #	1	1
	K/A #	026.A3.1	
	Importance rating	4.3	4.5

Proposed Question:

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

T + 2 minutes: Containment pressure is 23 psia and increasing.
T + 25 minutes: The crew is preparing to transition from E-0, *Reactor Trip or Safety Injection.*
T + 25 minutes: In preparation for the transition brief, the RO is comparing the current status of the CTMT Spray System components to their status prior to the event.

What changes (between the initial 100% power status and their current status), if any, are expected while verifying the status of the CTMT Recirculation Spray (RSS) Pumps and the QSS Pump Discharge Valves?

	RSS Pumps (3RSS*P1A-D)	QSS Discharge Valves (3QSS*MOV34A/B)
a)	Started	Remained Open
b)	Started	Stroked from Closed to Open
c)	Remained Off	Remained Open
d)	Remained Off	Stroked from Closed to Open

Proposed Answer: D

Explanation (Optional): On a CDA (23 psia CTMT pressure) the QSS Pumps immediately start and their discharge MOVs stroke open from the closed position. The RSS Pumps will not start on a CDA until the RWST Lo-Lo setpoint is reached, to ensure adequate CTMT sump level to minimize sump blockage concerns for RSS Pump operation. This takes a minimum of about 35-40 minutes after event initiation.
"A" and "B" are wrong, since the RSS Pumps do not automatically start until a minimum of about 35-40 minutes into the event, and currently, it has been 25 minutes since event initiation.
"A" and "B" are plausible, since the RSS pumps also have an 11 minute timer interlock to allow automatic starting, and it has been 25 minutes since event initiation.
"C" is wrong, and "D" correct, since the QSS valves are normally maintained closed per Tech Specs for CTMT Isolation requirements, and they stroke open on a CDA. "C" is plausible, since the RSS discharge valves are normally open, and the QSS valves are required to be open for QSS Spray Pump operation.

Technical Reference(s): OP 3353.MB2B (Rev. 0), 1-8
(Attach if not previously provided, including version/revision number.) LSKs 24-9.4A (Rev. 12), 24-9.4B (Rev. 12), and 24-9.4Q (Rev. 9)
LSK 27-11J (Rev. 11)
P&IDs 112C (Rev. 38) and 115A (Rev. 37)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following containment de-pressurization system components controls and interlocks… QSS pump discharge isolation valves (QSS*MOV34A, B)… Containment recirculation pumps…

Question Source: Bank #402560
Question History: Last NRC Exam 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 # 42	Tier #	2	2
K/A Statement: Effect of loss or malfunction of the	Group #	1	1
CTMT spray system on recirculation spray system	K/A #	026.K3.2	
Proposed Question:	Importance Rating	4.2	4.3

The plant has been operating at 100% power when a large break LOCA occurs, resulting in the following sequence of events:

T=0:	The Reactor trips and Safety Injection actuates.
T+6 minutes:	CDA actuates.
T+6 minutes:	Both Quench Spray pumps FAIL to automatically or manually start.
T+16 minutes:	The crew transitions to EOP 35 FR-Z.1, *Response to High Containment Pressure*.
T+17 minutes:	A PEO is dispatched to the ESF building to realign RSS pump "C", using FR-Z.1, Attachment "B", "Local Alignment of Recirculation Spray Pump".

Complete the following statement concerning the specific action the PEO will take with RWST Recirculation Suction Valves 3QSS*AOV27 and 28, and to where the PEO will align the "C" RSS Pump Suction path.

Per FR-Z.1, Attachment "B", the PEO will block 3QSS*AOV27 and 28__(1)__, and align the "C" RSS pump to take a suction on the __(2)__.

	(1)	(2)
a)	CLOSED to isolate the non-safety grade portion of the system	RWST
b)	CLOSED to isolate the non-safety grade portion of the system	CTMT Sump
c)	OPEN to provide a flowpath	RWST
d)	OPEN to provide a flowpath	CTMT Sump

Proposed Answer: C

Explanation (Optional): With no Quench Spray Pump running, steam is not being condensed as effectively in CTMT, and RWST water is not being sprayed into CTMT. So CTMT sump level may be inadequate for taking a suction on the CTMT sump, and CTMT pressure is not lowering as quickly as designed, extending radiation release through the CTMT boundary.
"A" and "B" are wrong, since Att. B of FR-Z.1 will direct the PEO to locally block OPEN the RWST Recirc Suction Valves to provide a flowpath for the RSS Pumps."A" and "B" are plausible, since these valves are designed to close on a SIS, to isolate the RWST recirculation heater loop from the RWST, to isolate the non-safety grade RWST heater loop piping from the RWST. Also, E-1 series procedures direct the operators to isolate the non-safety grade Charging Pump min-flow recirculation path to the RWST.
"C" is correct, and "D" wrong, since Attachment B of FR-Z.1 will align the selected RSS pump to take a suction on the RWST, effectively converting the selected RSS pump into a QSS pump. "D" is plausible, since the RSS System is designed to take a suction on the CTMT Sump during an accident, and would be correct if Quench Spray were functioning. Also, RCS inventory has been discharged to CTMT.

Technical Reference(s): FR-Z.1 (Rev. 17), Att. B, pages 1 and 2
(Attach if not previously provided, including version/revision number.) P&ID 112C (Rev. 38)
P&ID 115A (Rev 37)

Proposed references to be provided to applicants during examination: None

Learning Objective: DESCRIBE the Major Action Categories within EOP 35 FR-Z.1

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.3, 41.8, and 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 43	Tier #	2	2
K/A Statement: Ability to locate control room switches, controls and indications, and to determine they correctly reflect the desired plant lineup for the main and reheat steam system	Group #	1	1
	K/A #	039.GEN.2.1.31	
	Importance rating	4.6	4.3

Proposed Question:

The crew is responding to a turbine trip per AOP 3550, *Turbine Trip*.

The BOP operator is monitoring the Main Boards for proper system response to the turbine trip.

What is the expected status of the Reheat Steam Supply Valves (3MSS-MOV50A/B) and the Reheat Steam Pressure Control Valves (3MSS-PDV36A/B)?

	Reheat Steam Supply Valves (3MSS-MOV50A/B)	Reheat Steam Pressure Control Valves (3MSS-PDV36A/B)
a)	OPEN	OPEN
b)	OPEN	CLOSED
c)	CLOSED	OPEN
d)	CLOSED	CLOSED

Proposed Answer: D

Explanation (Optional):
This question is based on Millstone 3 OE, when on a turbine trip in 2016, one of the two Reheat Steam Supply Valves failed to close. This resulted in excessive cooldown, and also some confusion for the operators as to which of the valves was in the proper position. The Reheat Steam supply path to the Moisture Separator Reheaters (MSRs) is normally supplied by manually opening the Reheat Steam Supply Valves (3MSS-MOV50A/B), and then throttled via the Reheat Steam Pressure Control Valves (3MSS-PDV36A/B) by an automatic control system using a ramped reheat steam supply flow versus turbine power. The reheat steam supply pressure control valves are automatically throttled based on turbine cross around pressure, between about 30% turbine load and 65% turbine load.
"A" and "B" are wrong, since on a turbine trip, 3MSS-MOV50A/B will automatically close as turbine load drops below 10%. "A" and "B" are plausible, since 3MSS-MOV 50A/B are normally open, and numerous secondary plant valves require manual realignment on a turbine trip per AOP 3550.
"C" is wrong, and "D" correct, since 3MSS-PDV36A/B will stroke closed as turbine load drops below about 30%. "C" is plausible, since 3MSS-PDV36A/B are normally fully open above 65% power, and numerous secondary plant valves require manual realignment on a turbine trip per AOP 3550.

Technical Reference(s): AOP 3550 (Rev. 10). Step 10.d
(Attach if not previously provided, including version/revision number.) OP 3317 (Rev. 24), steps 4.6.1 through 4.6.4
LSK 25212-28533 Sheet 2 (Rev. 9)
P&ID 123C (Rev. 23)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following moisture separator reheater system components controls and interlocks: a. Steam supply valves (MSS-MOV50A/B)
b. Steam load control valves (MSS-PDV36A/B AND PDV37A/B)…

Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4, 41.7, and 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 44	Tier #	2	2
K/A Statement: Main and reheat steam system design feature or interlock which provides for reactor building isolation	Group #	1	1
	K/A #	039.K4.7	
	Importance Rating	3.4	3.7

Proposed Question:

The plant is initially at 100% power.

A steamline break occurs in the Main Steam Valve Building, resulting in a Main Steam Isolation (MSI) signal.

Which one of the following valves receives an automatic CLOSE signal directly from the MSI signal?

a) Main Steamline Upstream Drain Valve 3DTM*AOV61A
b) Blowdown Ctmt Isolation Valve 3BDG*CTV22A
c) Atmospheric Relief Isolation Valve 3MSS*MOV18A
d) Atmospheric Relief Bypass Valve 3MSS*MOV74A

Proposed Answer: A

Explanation (Optional): This question is considered a KA match since per the EAL Tables, the Containment Barrier is considered "Lost" if there is an unisolable steam break outside Containment with SG tube leakage. The MSI signal is designed to isolate paths directly from the SGs.
On an MSI, the following valves receive an automatic CLOSE signal:

- The MSIVs
- The MSIV Bypass Valves
- The Atmospheric Relief Valves
- The MSS Upstream Drain Valves
- The TDAFW Pp Steam Supply Upstream Drain Valves

"A" is correct, since 3DTM*AV81A automatically closes on a MSI signal.
"B" is wrong, since 3BDG-HV20A does not automatically close on a MSI signal.
"C" is wrong, since MSS*MV18A does not automatically close on a MSI signal.
"D" is wrong, since 3MSS*MV74A does not automatically close on a MSI signal.
"B", "C", and "D" are plausible, since all of these valves are in paths that when closed, isolate a flowpath out of the Steam Generator, which is faulted outside of CTMT. Also, the correct answer is not one of the main valves that automatically isolate, such as the MSIVs, MSIV Bypass Valves, or Atmospheric Relief Valves.

Technical Reference(s): Functional Sheet 8 (Rev. K)
(Attach if not previously provided, including version/revision number.) P&IDs 123A (Rev. 61), 123B (Rev. 26), 123D (Rev. 14)
P&ID 123E (Rev. 25), 145A (Rev. 44)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the Main Steam System under the following… Receipt of a Main Steam Isolation actuation signal
Question Source: Bank #403956
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 45	Tier #	2	2
K/A Statement: Main Feedwater design feature or interlock which provides for automatic turbine/reactor trip runback	Group #	1	1
	K/A #	059.K4.2	
	Importance Rating	3.3	3.5

Proposed Question:

The plant is at 100% power when a feedwater transient occurs:

The crew determines the cause of the transient is a Main Turbine runback.

Which abnormal condition in the Stator Coolant (GMC) System could **NOT** be the cause of the runback?

a) High temperature
b) High conductivity
c) Low inlet pressure
d) Low flow versus generator current

Proposed Answer: B

Explanation (Optional):
A turbine runback and energization of stator coolant protection circuitry occurs on high stator coolant temperature, low stator cooling inlet pressure, and high current to flow comparison.
"A" is wrong, since high temperature inputs to the runback circuit.
"B" is correct, since high stator coolant conductivity does not input into the stator coolant runback circuit.
"C" is wrong, since low inlet pressure inputs to the runback circuit.
"D" is wrong, since the current to flow comparator inputs to the runback circuit.
"A", "C", and "D" are plausible, since high conductivity is a major concern. Stator coolant is pumped directly through the stator bars, and de-ionized water is used to prevent creating electrical shorts in the stator bars. High stator coolant conductivity requires a manual turbine trip when above 9.9 umho/cm.

Technical Reference(s): OP 3324E (Rev. 13), Precaution 3.12
(Attach if not previously provided, including version/revision number.) OP 3353.SCP (Rev. 3), 2-2
OP 3353.MB7C (Rev. 13), 3-3A, Setpoint and 2.1

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the EHC Control System under the following normal, abnormal, and emergency conditions… Stator Coolant System initiated Main Turbine RUNBACK
OT(DELTA)T initiated Main Turbine RUNBACK
OP(DELTA)T initiated Main Turbine RUNBACK
Power Load Unbalance initiated Main Turbine RUNBACK…

Question Source: Bank #403639
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 46	Tier #	2	2
K/A Statement: Predict/monitor changes in SG level due to operating auxiliary feedwater system controls	Group #	1	1
	K/A #	061.A1.1	
Proposed Question:	Importance Rating	3.9	4.2

The plant is in HOT STANDBY, and initial conditions are as follows:

- Tave is 557°F.
- SG levels are being maintained using both MDAFW Pumps in their normal alignment.
- "A", "C" and "D" SG NR levels indicate 50% and stable.
- "B" SG NR Level indicates 40% and stable.

The following sequence of events occurs:

1. The BOP operator throttles open on the AFW flow control valve to the "B" SG, to raise "B" SG NR level back to 50%.
2. The BOP monitors both Wide Range and Narrow Range levels for the "B" SG while raising "B" SG NR level back to 50%.
3. When "B" SG NR level indicates 50%, the BOP operator throttles the "B" AFW flow control valve back to its original position.

Complete the following statements concerning "B" SG level response due to the BOP operator's actions.

Shortly after commencing feeding the "B" SG at an increased rate, ___(1)____level indication initially provides the best indication of the actual change in mass in the "B" SG level.
After the BOP operator throttles the "B" AFW flow control valve back to its original position, ___(2)____level indication may continue increasing for a short while.

	(1)	(2)
a)	Narrow Range	Narrow Range
b)	Narrow Range	Wide Range
c)	Wide Range	Narrow Range
d)	Wide Range	Wide Range

Proposed Answer: C

Explanation (Optional): "A" and "B" are wrong, since when the AFW flow is initially increased, cold AFW water will cause shrink in NR level indications, as it actually monitors downcomer level, which is affected by the boiling rate in the tube bundle region. "A" and "B" are plausible, since Narrow Range and Wide Range levels are monitoring SG level, and this is how wide range level will respond.
"C" is correct, since wide range level taps are at the bottom and top of the SG, measuring actual mass in the SG, independent of NR shrink and swell problems.
"D" is wrong, since when level reaches 50%, the cold water that has been added to the SG will continue to heat up, with thermal expansion raising indicated NR levels for a time after AFW flow is reduced to its original value, "D" is plausible, since restoring AFW to its original value restores the flowrate required to maintain SG inventory stable, and this is how wide range level will respond.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP 3203 (Rev. 26), Caution prior to step 4.2.8 OP 3322 (Rev. 32), Precaution 3.6 P&ID 130B (Rev. 49) P&ID 130C (Rev. 27) GFS

Proposed references to be provided to applicants during examination: None

Learning Objective: DESCRIBE the major administrative & procedural precautions & limitations placed on the operation of the Auxiliary Feedwater System, & the basis for each.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4 and 41.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 47	Tier #	2	2
K/A Statement: Electrical power supply for the electric driven auxiliary feedwater pumps	Group #	1	1
	K/A #	061.K2.2	
Proposed Question:	Importance Rating	3.7	3.7

Initial conditions:

- The plant is at 30% power.
- The "A" MDAFW Pump (3FWA*P1A) is running as part of a retest.

The Main Turbine trips.

Assuming no operator actions have been taken, complete the following statement describing the path offsite power is taking to supply power to 3FWA*P1A.

Power is being supplied from the 345KV Switchyard_____.

a) through the "A" Normal Station Service Transformer (NSSA) via the Main Transformers
b) directly through the "A" Normal Station Service Transformer (NSSA)
c) directly through the "A" Reserve Station Service Transformer (RSSA), due to a fast transfer
d) directly through the "A" Reserve Station Service Transformer (RSSA), due to a slow transfer

Proposed Answer: A

Explanation (Optional):
Prior to the turbine trip, the electrical path from the generator output breaker to the grid was through the Main Transformers, and the electrical path to the 4KV busses was directly through NSST A, which received power from the Main Generator output via a tap between the Generator Output Breaker and the Main Transformers. On a turbine trip below 51% power, the reactor remains on line, but the Main Generator trips and the Generator Output Breaker opens.
"A" is correct, and "B" wrong, since offsite power will now backflow through the Main Transformers, to NSST "A", to 4KV bus 34C, to the "A" AFW Pump.
"B" is plausible, since originally, power went directly from the generator output to NSST "A". Also, if the RSST becomes the power source, this receives power directly from offsite power.
"C" and "D" are wrong, since no transfers occur, since offsite power back-feeds through the Main Transformers to the NSSTs.
"C" and "D" are plausible, since if the NSST is lost, a fast or slow transfer occurs, depending specific conditions. Also, for plants without Main Generator output breakers, a transfer occurs to the RSST whenever the main generator trips.

Technical Reference(s): FSAR (Rev. 31.2) Section 8.3.1.1.1
(Attach if not previously provided, including version/revision number.) EE-1A (Rev. 26)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the 4 kV Distribution System operation under normal, abnormal, and emergency conditions… Main Generator trip…
Question Source: Modified Bank #402120 (Parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7

Comments:
The question is considered "modified", since in the stem power level has been changed from 100% to 30%, meaning the reactor no longer trips. The question also has been changed to specifically address a running AFW Pump. One distractor that listed the EDG as the power source has been changed to the NSST being the source.

Original question:

The plant is initially at 100% power in a normal electrical line-up.

The Main Turbine trips.

What is the expected result for the 4160VAC buses?

a) The Emergency Diesel Generators start and close onto 4KV emergency busses 34C and 34D. Non-emergency buses 34A and 34B are de-energized (supply and tie breakers open).
b) A fast-transfer occurs, resulting in the Reserve Station Service Transformer (RSSA) supplying all four 4KV buses.
c) A slow-transfer occurs, resulting in the Reserve Station Service Transformer (RSSA) supplying 4KV emergency buses 34C and 34D. Non-emergency buses 34A and 34B are de-energized (supply and tie breakers open).
d) Offsite power is back-fed through the Main Transformers and continues to supply all four 4KV buses through the Normal Station Service Transformer (NSSA) with no interruption of power.

Correct Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Question # 48	Tier #	2	2
K/A Statement: Electrical power supply for major system loads for the AC electrical distribution system	Group #	1	1
	K/A #	062.K2.1	
Proposed Question:	Importance Rating	3.3	3.4

Which of the following plant loads are powered from 6.9KV Bus 35C?

a) Main Feedwater Pump 3FWS-P1
b) Condensate Pump 3CNM-P1C
c) Circulating Water Pump 3CWS-P1C
d) Heater Drain Pump 3HDL-P1C

Proposed Answer: A

Explanation (Optional):
A" is correct, since the loads powered from 35C are the Reactor Coolant Pump 3RCS-P1C and Main Feedwater Pump 3FWS-P1.
"B" is wrong, since Condensate Pump 3CNM-P1C is powered from 6.9KV Bus 35D.
“B” is plausible, since 3CNM-P1C is a 6.9KV load, with “C” as its designator.
"C" is wrong, since 3CWS-P1C is powered from Bus 34A.
“C” is plausible, since 3CWS-P1C is a large non-emergency load (4.16KV) with “C” as its designator.
"D" is wrong, since 3HDL-P1C is powered from Bus 34A.
“D” is plausible, since 3HDL-P1C is a large non-emergency load (4.16KV) with “C” as its designator.

Technical Reference(s): EE-1F (Rev. 21)
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: DESCRIBE the function and location of the following major 6 KV system components… 6 KV switchgear…
Question Source: Bank #402163
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 49	Tier #	2	2
K/A Statement: Predict impact/mitigate	Group #	1	1
grounds on the DC electrical system	K/A #	063.A2.1	
Proposed Question:	Importance Rating	2.5	3.2

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

1. The Primary Rounds PEO reports indication of a "hard ground" on DC Bus 5.
2. The crew enters AOP 3551, *DC Bus Ground*.

Assuming all other conditions are normal, how is the ground indicated at DC Bus 5 (3BYS-PNL-5); and what is one strategy used in AOP 3551 to mitigate the event?

a) One of three indicating lights indicates significantly dimmer than the other two. AOP 3551 attempts to isolate the ground in a prescripted order, starting with the individual loads powered by the DC panels, and then moving to the equipment that has a greater impact on the plant when removed from service, such as the Battery Charger.
b) One of three indicating lights indicates significantly dimmer than the other two. AOP 3551 provides flexibility to perform the individual ground isolation steps out of order based on equipment that is suspected to be the source of the ground.
c) When a pushbutton is depressed, >110VDC difference exists between two meters. AOP 3551 attempts to isolate the ground in a prescripted order, starting with the individual loads powered by the DC panels, and then moving to the equipment that has a greater impact on the plant when removed from service, such as the Battery Charger.
d) When a pushbutton is depressed, >110VDC difference exists between two meters. AOP 3551 provides flexibility to perform the individual ground isolation steps out of order based on equipment that is suspected to be the source of the ground.

Proposed Answer: D

Explanation (Optional): Since the system is ungrounded, a single ground will not cause any breaker trips. With two or more grounds on a given bus, an overcurrent condition could exist, causing the faulty load to trip. So, it is important to detect the ground and isolate it in a timely fashion.
"A" and "B" are wrong, since ground detection is provided via a switch on the DC Bus panel, which places two meters in service. With a single hard ground on the system, there will be >110VDC difference between the grounded and ungrounded side of the system. "A" and "B" are plausible, since for the 480 VAC system, three white lights are mounted locally at the load center, and with a ground, one of the three lights will differ in brightness from the other two.
"C" is wrong, and "D" correct, since AOP 3551 provides the crew flexibility to perform steps out of order based on suspicion of the cause of the ground. Also, the step sequence in AOP 3551 starts with the Battery Charger, and ends with checking the individual loads. "C" is plausible, since there is a sequence of steps for isolating the ground in AOP 3551, and general rules of usage in AOPs and EOPs require the steps to be performed in order.

Technical Reference(s): AOP 3551 (Rev. 5), step 2
(Attach if not previously provided, SP 3670.1-009 (Rev. 11), page 15
including version/revision number.) EE-1BD (Rev. 23)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the major action categories within AOP 3551.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.5, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 50	Tier #	2	2
K/A Statement: Physical connections and/or cause/effect between the DC electrical system and battery charger and battery	Group #	1	1
	K/A #	063.K1.3	
	Importance Rating	2.9	3.5

Proposed Question:

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

1. Battery Charger 1 Output Breaker inadvertently trips open.
2. 10 minutes later, the BOP checks DC Bus 1 voltage.

Assuming no operator actions have been taken, what voltage will the BOP observe?

a) 0 volts
b) 125 volts
c) 135 volts
d) 140 volts

Proposed Answer: B

Explanation (Optional):
The DC Busses are normally supplied by two possible sources, the Battery Charger, which normally puts out 135VDC, and the Battery, which puts out 125VDC. This voltage difference maintains the battery on a float charge. The DC Bus then provides backup power to the Inverter, which supplies the Vital AC Instrument Bus (VIAC). The Inverter is also supplied by a rectifier, which provides 140 VDC from a 480 Volt Motor Control Center (MCC). The reason the Rectifier doesn't supply the DC Bus, even though it is at a higher voltage than the Battery Charger, is there is a reverse-biased blocking diode in the path. "A" is wrong, and "B" is correct, since on a loss of the charger, the DC bus will still be energized by the battery, which puts out 125 volts. "A" is plausible, since the source that normally provides power to the DC Bus has been lost.
"C" is wrong, since the Charger output breaker has opened. "C" is plausible, since this is the voltage normally indicated on the DC Bus with the Charger aligned.
"D" is wrong, since the blocking diode prevents the rectifier from supplying the DC Bus. "D" is plausible, since this is the voltage put out by the rectifier from the 480 VAC bus to the inverter.

Technical Reference(s): OP 3345C (Rev. 16-10), Sections 1.1, 2.1.2, and 2.1.3
(Attach if not previously provided, including version/revision number.) EE 1BA (Rev. 31)
Proposed references to be provided to applicants during examination: None
Learning Objective: Given a failure of the 125 VDC distribution system or a portion of the system, DETERMINE the effects on the system and on interrelated systems
Question Source: Bank #401948
Question History: Last NRC Exam Millstone 3 2011 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 51	Tier #	2	2
K/A Statement: Predict impact/mitigate the consequences of opening/closing breaker between buses (VARS, out of phase, voltage) on the EDG system	Group #	1	1
	K/A #	064.A2.8	
	Importance Rating	2.7	3.1

Proposed Question:

The BOP operator is manually transferring 4KV Bus 34C from the "A" EDG to RSST A per OP 3343, *Station Electrical Service 4.16KV*, and current conditions are as follows:

- The BOP is preparing to shift the "A" EDG from "Unit" to" Parallel" operation.
- Per OP 3343, the BOP adjusts EDG frequency to 60.8 hertz prior to selecting "Parallel".

What is the potential adverse consequence of failing to adjust frequency prior to selecting "Parallel" on the EDG Mode Selector Switch?

a) Automatic tripping of the EDG output breaker
b) Stripping Bus 34C on an inadvertent LOP signal
c) Tripping the Diesel on overspeed
d) Reverse-powering the diesel

Proposed Answer: A

Explanation (Optional):
Switching from "UNIT" to "PARALLEL" places speed droop in service for the diesel. This is done in preparation for operating the diesel in parallel with another AC source, allowing the faster source to slow down to share load with the other source.
"A" is correct, since placing speed droop in service will cause diesel generator frequency to decrease when supplying a bus by itself. If frequency decreases to 59.0 Hz the diesel output breaker will trip.
"B" is wrong, since if a LOP occurs due to the EDG output breaker tripping, bus stripping is desired to prevent overloading the oncoming power source. "B" is plausible, since the LOP signal occurs on lowered bus voltage.
"C" is wrong, since selecting Unit will cause the EDG to slow down, not speed up. "C" is plausible, since selecting unit or parallel affects EDG speed (frequency).
"D" is wrong, since selecting parallel while in unit risks the EDG picking up too much load, not too little.
"D" is plausible, since selecting unit when in parallel with another source results in diesel frequency decreasing, but this is due to excess load.

Technical Reference(s): OP3343 (Rev. 15), step 4.2.4.c, including Caution
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Emergency Diesel Generator System (including Support Systems), and the basis for each.
Question Source: Bank #403129
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 52	Tier #	2	2
K/A Statement:	Group #	1	1
Knowledge of LCOs and safety limits for the	K/A #	073.GEN.2.2.22	
process radiation monitoring system	Importance Rating	3.8	4.0

Proposed Question:

Which of these Radiation Monitors is a TRM required release path monitor?

a) 3ARC21, Air Ejector Exhaust Monitor
b) 3HVR10A, Turbine Building Stack Monitor
c) 3GWS48, Gaseous Waste Discharge Monitor
d) 3HVR18, Waste Building Ventilation Exhaust Monitor

Proposed Answer: B

Explanation (Optional):
The rad effluent release path monitors have been moved from Tech Specs to the TRM. The two release path rad monitors specified in the TRM are 3HVR10A and 3HVR19A.
"A" is wrong, since ARC21 is not specified as required per the TRM. "A" is plausible, since ARC21 is a process rad monitor that monitors a release path from Condenser Air Ejectors, to the Millstone stack.
"B" is correct, since HVR10A is a required release path process rad monitor per the TRM.
"C" is wrong, since GWS48 is not specified as required per the TRM. "C" is plausible, since GWS48 is a process rad monitor that monitors a release path from the Gaseous Waste System to the Millstone stack.
"D" is wrong, since HVR18 is not specified as required per the TRM. "D" is plausible, since HVR18 process rad monitor that monitors a release path from the Waste Building ventilation system to the Millstone stack.

Technical Reference(s): TRM 3.3.3.6.1 (LBDCR 07-MP3-018)
(Attach if not previously provided, P&ID 148A (Rev. 42)
including version/revision number.) P&ID 148D (Rev. 18)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Radiation Monitoring System, including the basis for each.
Question Source: Bank #404664
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5, 41.11, 43.2
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 53	Tier #	2	2
K/A Statement:	Group #	1	1
Physical connections or cause/effect relationship between the service water system and the diesel generator	K/A #	076.K1.5	
	Importance Rating	3.8	4.0

Proposed Question:

A loss of All AC power occurs, resulting in the following sequence of events:

1. The crew enters ECA-0.0, *Loss of All AC Power*.
2. A PEO is dispatched to locally start the "B" EDG using ECA-0.0, Attachment E.
3. The PEO at the diesel discovers that Diesel Service Water Outlet Valve 3SWP*AOV39B is CLOSED.

Per ECA-0.0, Attachment E, what action, if any, is required to be performed in the “B” EDG enclosure to open 3SWP*AOV39B?

a) Align air to the valve operator using the manual air supply valve on the ground level.
b) Vent air directly from the valve operator on the mezzanine.
c) Vent air using the manual air vent valve on the ground level.
d) No local action is required, since the valve will open when the diesel energizes the bus.

Proposed Answer: B

Explanation (Optional):

"A" is wrong, since the valve fails open on loss of air. “A” is plausible, since several valves require air to open. PEOs align nitrogen to operate the Atmospheric Relief Valves in ECA-0.0.
"B" is correct, since Attachment E, step 1 has the operator open the service water valve by venting air directly off of the 3SWP*AOV39B valve operator.
"C" is wrong since for the “B” EDG, there is no vent valve. “B” is plausible, since this physical connection is available on the “A” EDG, and this is the action required for 3SWP*AOV39A.
"D" is wrong, since the LOP should have already opened the valve, and this is a contingency action if the valve has not automatically opened. “D” is plausible, since many components do not operate with no AC power, and several automatic actions occur (such as load sequencing) based on power being restored.

Technical Reference(s): ECA-0.0 (Rev. 35) Att. E, step E.1 RNO
(Attach if not previously provided, including version/revision number.) P&ID 133D (Rev. 53)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following Service Water System components… EDG Service Water Outlet Valves (SWP*AOV39A/B)…

Question Source: Bank #407616

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.4, 41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 54	Tier #	2	2
K/A Statement: Knowledge of the effect a loss or malfunction of the instrument air system will have on systems having pneumatic valves and controls	Group #	1	1
	K/A #	078.K3.2	
	Importance Rating	3.4	3.6

Proposed Question:

Current conditions:

- A complete loss of instrument air pressure has occurred.
- The crew has just entered ES-0.1, *Reactor Trip Response.*

Complete the following statement concerning options available to the operators in the Control Room while in this condition.

Letdown can be aligned through the head vent path to the___(1)___, and RCS pressure can be reduced via the___(2)___.

	(1)	(2)
a)	VCT	Auxiliary Spray Valve
b)	VCT	PORVs
c)	PRT	Auxiliary Spray Valve
d)	PRT	PORVs

Proposed Answer: D

Explanation (Optional):
This question tests a point of confusion experienced by the operators during the Millstone 3, May 35, 2014 LOP event concerning the availability of excess letdown to the VCT on a loss of air (OE).
"A" and "B" are wrong, since the Excess Letdown Flow Control Valve to the VCT (3CHS-HC123) fails closed on a loss of instrument air. "A" and "B" are plausible, since on a loss of instrument air, letdown isolates and the excess letdown path to the VCT is normally available as a backup letdown path.
“C” is wrong, since the Auxiliary Spray Valve (3RCS*AV8145) fails closed on a loss of instrument air.
“C” is plausible, since on a loss of instrument air, the normal Pzr Spray Valves fail closed, and the Auxiliary Spray Valve is normally available as a backup to the Pzr Spray Valves.
"D" is correct, since both the head vent path to the PRT and the Pzr PORVs are available on a loss of instrument air pressure.

Technical Reference(s): AOP 3562 (Rev. 17), Attachment C, page 1
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss the basis of major precautions, procedure steps, or sequence of steps.
Question Source: Bank #408737
Question History: Last NRC Exam Millstone 3 2000 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4, 41.7, 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 55	Tier #	2	2
K/A Statement: Physical connections and/or cause/effect relationship between the containment system and containment isolation/containment integrity	Group #	1	1
	K/A #	103.K1.2	
	Importance Rating	3.9	4.1

Proposed Question:

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

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

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

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

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

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

Proposed Answer: C

Explanation (Optional):
"A" is wrong, since the steam pressure high rate signal provides an MSI signal, but not an SIS signal.
"A" is plausible, since the low steam pressure signal provides a SIS and MSI, but this was manually blocked by the crew when RCS pressure dropped below P-11 (2000 psia).
"B" is wrong, since during the plant cooldown, the crew blocked the low Pzr pressure SIS signal per OP 3208 when RCS pressure dropped below P-11 (2000 psia). "B" is plausible, since prior to blocking this signal, Pzr pressure below 1892 psia would have generated a SIS signal.
"C" is correct, since the CTMT Hi 1 pressure SI signal is still in service, actuating SIS at 18 psia.
"D" is wrong, since below P-11, the crew also blocked the low steamline pressure SIS/MSI, and the high steam pressure rate MSI was instated. "D" is plausible, since prior to blocking this signal, Main Steamline pressure below 660 psig would have generated a SIS signal.

Technical Reference(s): E-0 (Rev. 32), Att. A, Sections A.1 and A.2
(Attach if not previously provided, including version/revision number.) OP 3208 (Rev. 31), step 4.2.5
Functional Sheets 6 (Rev. J) and 8 (Rev. K)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the following RPS controls and interlocks… Safety Injection a) Manual b) Ctmt Hi-1 c) Lo Steam Line Pressure d) Pressurizer Low Pressure…

Question Source: Bank #404943

Question History: Last NRC Exam Millstone 3 2009 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.4, 41.7, 41.9

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 56	Tier #	2	2
K/A Statement: Ability to manually operate or monitor determination of SDM from the control room	Group #	2	2
	K/A #	001.A4.11	
Proposed Question:	Importance Rating	3.5	4.1

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

- The crew is preparing to startup the reactor per OP 3202, *Reactor Startup*.
- The crew is preparing to withdraw Shutdown Bank A rods.
- The BOP operator is performing a SHUTDOWN MARGIN calculation in preparation for withdrawing the Shutdown Bank A rods per OP 3209B, *Shutdown Margin*.

When determining how much boron is required in the RCS per OP 3209B, the BOP is required to_____.

a) use the "Shutdown Banks Out" curve, in order to provide increased Shutdown Margin due to the potential for a stuck rod after the shutdown banks are withdrawn
b) use the "Shutdown Banks Out" curve, in order to provide sufficient "shutdown amount" to ensure an inadvertent MODE change does not occur during the withdrawal of the shutdown bank rods
c) interpolate between the “All Rods In” curve and the “Shutdown Banks Out” curve, to prevent excessive boration and provide increased Shutdown Margin due to the potential for a stuck rod after Shutdown Bank A rods are withdrawn
d) interpolate between the “All Rods In” curve and the “Shutdown Banks Out” curve, to prevent excessive boration and provide sufficient "shutdown amount” to prevent an inadvertent MODE change during the withdrawal of the Shutdown Bank A rods

Proposed Answer: B

Explanation (Optional):
"A" is wrong, since a stuck rod is already assumed before rods are withdrawn, so sufficient boron already exists before Shutdown Banks are withdrawn. “A” is plausible, since a rod sticking out is not possible when all rods are on the bottom.
"B" is correct, since withdrawing shutdown banks decreases shutdown amount. Per Tech Spec definitions, MODE 2 is differentiated from MODE 3 by Keff greater than or equal to 0.99 per Tech Spec definitions. But the crew is not able to monitor Keff, so the transition from MODE 3 to MODE 2 is procedurally called after shutdown rods have been withdrawn and control banks are starting to be withdrawn. Also, operators are not allowed to interpolate between curves on a graph.
"C" and “D” are wrong since operators are not allowed to interpolate between curves on a graph while determining Shutdown Margin boron requirements. “C” and “D” are plausible, since the operator is performing the Shutdown Margin Calculation prior to withdrawing Shutdown Bank A rods, and interpolating would prevent over-borating the RCS.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
Curve RE-B-02 (MP3-18-00, Rev. 0)
OP 3209B (Rev. 11), Sections 1.1 and 1.2
OP 3209B (Rev. 11), Precautions 3.2 and 3.3
OP 3202 Rev. 24), steps 2.4.7, 4.9, 4.10.11, 4.26.3, and 4.26.5

Proposed references to be provided to applicants during examination: None

Learning Objective: State why there is more than one curve for expected rod position and temperature.

Question Source: Bank #406725

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.1, 41.6, 41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 57	Tier #	2	2
K/A Statement: Electrical power supply for the Pzr heaters	Group #	2	2
	K/A #	011.K2.2	
Proposed Question:	Importance Rating	3.1	3.2

Which Load Center provides power for Pressurizer Backup Heater Group "E"?

a) 32E
b) 32N
c) 32S
d) 32V

Proposed Answer: B

Explanation (Optional):
"B" is correct, and "A", "C", and "D" wrong, since Pzr Backup Heater Group "E" is powered from non-emergency Load Center 32N.
"A" is plausible, since 32E is a non-emergency load center at Millstone 3.
"C" is plausible, since 32S is an emergency load center that supplies power to Backup Heater Group "A"
"D" is plausible, since 32V is an emergency load center that supplies power to Backup Heater Group "B"

Technical Reference(s): OP 3301G-001 (Rev. 5-3), 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: Describe the function and location of the following major 480 volt AC system components… 480 volt Load Centers…
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 58	Tier #	2	2
K/A Statement: Effect of loss or malfunction of the NIS system on CRDS	Group #	2	2
	K/A #	015.K3.2	
Proposed Question:	Importance Rating	3.3	3.5

Reactor Power is initially stable at 90% with Control Bank D rods at 210 steps.

The following sequence of events occurs:

1. The crew enters AOP 3575, *Rapid Downpower.*
2. The RO commences borating the RCS.
3. The BOP is preparing to commence reducing power at 5% per minute.

The crew is unaware that Power Range NIS channel N-44 has failed as-is at 90% power.

Assuming no operator action is taken to address the instrument failure and Rod Control is left in automatic, complete the statements below concerning how the failed NIS channel will affect Rod Control as the downpower is commenced,

Initially, control rods will drive inward at a___(1)___ rate than expected, and the reason Control Rods eventually stop driving in is the___(2)___.

	(1)	(2)
a)	faster	Temperature Error signal eventually exceeds the Power Mismatch signal
b)	faster	Power Mismatch signal eventually exceeds the Temperature Error signal
c)	slower	Temperature Error signal eventually exceeds the Power Mismatch signal
d)	slower	Power Mismatch signal eventually exceeds the Temperature Error signal

Proposed Answer: A

Explanation (Optional): Automatic Rod Control uses two signals to determine rod speed. A Power Mismatch circuit compares the rate of change between Primary power (based on auctioneered high NIS Power) and secondary power (based on selected Turbine impulse pressure channel). This signal causes rods to start driving in anticipation of upcoming RCS temperature changes based on primary to secondary heat imbalance. The other signal is a temperature error signal, which compares Auctioneered High Tave with program Tave, which is based on Turbine power, based on Selected Turbine Impulse pressure.
"A" is correct, since with NIS (auct hi) failed as is, and secondary power decreasing, rods will start driving in more rapidly than expected due to the power mismatch signal. Tref is responding properly as turbine power decreases, and with rods driving in faster than expected, plus the effect of boration eventually assisting rods in lowering actual Tave, inward rod motion will stop as Temperature Error signal exceeds the Power Mismatch signal.
"B" is wrong, since Power Mismatch signal decays away as the downpower stops since it is based on rate of change, so Temperature Error signal will become dominant. And since rods have driven in further than expected due to the Power Mismatch signal, the Temperature error signal will resist further inward rod motion. "B" is plausible, since there will be a growing difference between sensed primary to secondary power with the failed NIS channel, but this is a rate of change signal.
"C" and "D" are wrong, since initially, with sensed primary power remaining steady, the Power Mismatch signal will be driving in rods at a more rapid rate. "C" and "D" are plausible, since this would be true if Tref was based on Reactor Power, and the temperature error signal is not affected by the NIS failure

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: For the following conditions, determine the effects on the Nuclear Instrumentation System and on interrelated systems… Power Range instrument failure in mode 1 above P-10…
Question Source: Bank #404791
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.6, 41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 59	Tier #	2	2
K/A Statement: Predict impact/mitigate the effect of a detector failure on the non-nuclear instrumentation system	Group #	2	2
	K/A #	016.A2.1	
	Importance Rating	3.0	3.1

Proposed Question:

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

The following initial sequence of events occurs:

1. Pressurizer pressure instrument 3RCS*PI458 (Channel 4) fails HIGH.
2. The crew enters AOP 3571, *Instrument Failure Response*.
3. The crew completes all actions of AOP 3571, including tripping all bistables.

RCS Loop A Tave instrument 3RCS*TI412 (Channel 1) fails HIGH due to its Thot narrow range temperature instrument failing HIGH.

Does the reactor automatically trip? If so, what signal caused the trip?

a) No. The plant remains at power.
b) Yes, due to an OTΔT reactor trip.
c) Yes, due to an OPΔT reactor trip.
d) Yes, due to a Pzr Hi Pressure reactor trip.

Proposed Answer: B

Explanation (Optional): "A" is wrong, and "B" is correct, since the original pressure transmitter failure's corrective action per AOP 3571 required the crew to trip the OTΔT bistable for loop "D" (Channel 4), along with other pressure-related bistables. And when the Thot channel failed high, sensed RCS channel 1 ΔT (Thot – Tcold) fails high, bringing in the channel 1 OTΔT bistable, meeting the 2 of 4 coincidence for an OTΔT trip. "A" is plausible, since the reactor trip coincidence for each of the above trips is 2 of 4 channel bistables in the tripped condition, and the two instruments that failed are not measuring the same type of parameter.
"C" is wrong, since the original failure did not require tripping the OPΔT bistable for Channel 4, since Pzr pressure does not feed into the OPΔT trip. "C" is plausible, since the second failure brings in an OPΔT bistable, causing 1 of 4 bistables to be lit for OPΔT.
"D" is wrong, since the second failure did not bring in a Pzr high pressure trip bistable. "D" is plausible, since the original failure required tripping the Pzr Hi Pressure Trip Bistable, causing 1 of 4 Pzr Hi Pressure trip bistables to be lit.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)
- AOP 3571 (Rev. 16), Attachment B, page 6 of 6
- AOP 3571 (Rev. 16), Attachment A, page 8 of 9
- Functional Sheets 5 (Rev. K) and 6 (Rev. J)
- P&ID 102A (Rev. 34)

Proposed references to be provided to applicants during examination: None

Learning Objective: DISCUSS conditions which require transition to other procedures (from AOP 3571).

Question Source: Bank #407244

Question History: Last NRC Exam Millstone 3 2011 NRC Exam

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5, 41.10, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 60	Tier #	2	2
K/A Statement: Effect of a loss or malfunction of the MSIVs on the steam generator system	Group #	2	2
	K/A #	035.K6.1	
Proposed Question:	Importance Rating	3.2	3.6

The plant is at 80% power when the "B" MSIV spuriously closes.

Assuming no operator action is taken, what effect will this have on the plant?

a) Reactor power stabilizes at 60% due to the loss of steam flow from the "B" Steam Generator.
b) Reactor power stabilizes at 80% with the "A", "C", and "D" SGs each providing more steam flow.
c) The reactor will trip on Lo-Lo NR level in the "B" SG.
d) The reactor will trip on High Pressurizer level.

Proposed Answer: C

Explanation (Optional):
"A", "B", and "D" are wrong, and "C" correct, since when the MSIV closes, "B" SG narrow range level will rapidly shrink to the LO-LO level trip setpoint. This is due to the loss of a steam release path out of the SG when the MSIV closes resulting in significantly less boiling in the SG, leading to the downcomer water flowing into the SG tube area.
"A" is plausible, since actual mass in the SG increases on the loss of steam flow, and at low powers, shrink is not as significant as at high powers, and power is below 100%. Also, 60% power is based on each of the four SGs providing 20% of the total steam flow.
"B" is plausible, since actual mass in the SG increases on the loss of steam flow, and at low powers, shrink is not as significant as at high powers, and power is below 100%. Also, if the reactor did not trip, the turbine control valves would remain in their original position, and the four SGs all supply a common steam header manifold prior to supplying the Main Turbine, and steam header manifold pressure will drop when one of the SGs no longer is supplying it.
"D" is plausible, since on a loss of heat removal from the primary plant (such as a turbine trip), the RCS will heat up, expanding the RCS water, causing RCS pressure and Pzr Level to increase.

Technical Reference(s): AOP 3554 Basis Document (Rev. 11), step 2 Basis
(Attach if not previously provided, including version/revision number.) Steam Generator System Training Text (Rev. 3, Ch. 1), page 26
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the following phenomenon associated with the operation of Steam Generators: A. Shrink B. Swell
Question Source: Bank #405196
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.4, 41.5, 41.7, 41.14
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 61	Tier #	2	2
K/A Statement: Predict/monitor changes in steam pressure due to operating steam dump system controls	Group #	2	2
	K/A #	041.A1.2	
Proposed Question:	Importance Rating	3.1	3.2

The crew is preparing to commence a plant cooldown in accordance with OP 3208, *Plant Cooldown*, and initial conditions are as follows:

- RCS Tave is at 557°F.
- The Condenser Steam Dump Valves are aligned as follows:
  - The Control Mode Selector Switch is selected to the "Steam Pressure" mode.
  - The Steam Dump Interlock Selector Switch is in the ON position.
  - The Steam Dump Controller is selected to MANUAL.

The following sequence of events occurs.

1. The BOP operator depresses INCREASE on the Steam Dump Controller, slowly raising output from 0% to 40%.
2. The BOP commences monitoring Main Steam Header pressure on 3MSS-PI507 to verify proper plant response.
3. Steam pressure starts decreasing, and when RCS Tave decreases to 553°F, pressure stops decreasing.

Complete the following statement concerning the action required with Steam Dump Controls to reinitiate the cooldown, and how Main Steam Header pressure will respond to this action.

After selecting___(1)___and adjusting Steam Dump Controller output back to 40%, steam pressure will again start decreasing at___(2)___what was observed before reaching 553°F.

	(1)	(2)
a)	"Bypass Interlock" on the Interlock Selector Switches	a slower rate than
b)	"Bypass Interlock" on the Interlock Selector Switches	the same rate as
c)	"Reset" on the Control Mode Selector Switch	a slower rate than
d)	"Reset" on the Control Mode Selector Switch	the same rate as

Proposed Answer: A

Explanation (Optional):
Above P-12, raising the controller output to 40% causes steam dump valves to open in a staggered fashion, with three valves throttling open from zero to 33% open, three more valves throttling open from 33% to 67% output, and the final three valves throttling open from 67% to 100% open. Since the controller was set to 40%, three valves were fully open, and three more valves were throttled open. Reaching P-12 (553°F) causes all steam dump valves to close, preventing an inadvertent cooldown in the event of a steam dump system failure.
"A" is correct, since in order to remove P-12 from the circuit, "Bypass Interlock" must be selected, at which point only the three cooldown valves will open, so steam pressure will be decreasing at a slower rate than before.
"B" is wrong, since only the three cooldown dumps will open below P-12, and prior to P-12, six of the nine steam dump valves were open or throttled open. "B" is plausible, since controller output has not changed, and this would be true if the controller setpoint was set to 33% or less.
"C" and "D" are wrong, since in order to remove P-12 from the circuit, "Bypass Interlock" must be selected. "C" and "D" are plausible, since this is an actual switch and actual switch position on Main Board 5 used for steam dump control, and "Reset" removes C-7 (Load Reject) from the circuit, not P-12.

Technical Reference(s): OP 3316A (Rev. 16), step 4.2.14
(Attach if not previously provided, OP 3208 (Rev. 31), steps 4.2.9.f and 4.2.9.g
including version/revision number.) Functional Sheet 10 (Rev. J), including Note 9
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the steam dump system (when in the steam pressure mode of operation) during the following… Plant cooldown operations…
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.5, 41.7
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 62	Tier #	2	2
K/A Statement: Operational implications of the relationship of hydrogen/oxygen concentrations to flammability as it applies to the waste gas disposal system	Group #	2	2
	K/A #	071.K5.4	
	Importance Rating	2.5	3.1

Proposed Question:

Current conditions:

- Work control is preparing a tagout that isolates a section of piping on the discharge of the Degasifier in the Gaseous Waste System in preparation for maintenance on the system.
- The work that will be done requires disassembly of manual valve 3GWS-V27 at the inlet of the "B" Process Gas Precooler in the Process Gas System.

In accordance with the precautions of OP 3337, *Gaseous Waste System*, what verification or action is required to be taken prior to commencing this work?

a) Log into Tech Spec LCO 3.6.6.1, SUPPLEMENTARY LEAK COLLECTION AND RELEASE SYSTEM.
b) Verify the work does not violate the Containment boundary per LCO 3.6.3, CONTAINMENT ISOLATION VALVES.
c) Purge the piping with Nitrogen due to the potential for flammable concentrations of Hydrogen and Oxygen.
d) Verify Stack Discharge Stop Valve 3HVR*V42 is open, to ensure the system is vented prior to hanging the tags.

Proposed Answer: C

Explanation (Optional):
"A" is wrong, since a portion of the system being isolated and open to the Aux Building will not violate the SLCRS boundary. "A" is plausible, since this is a misapplication of Precaution 3.3, which requires a fan running or the stack isolation valve closed while SLCRS is required to be OPERABLE.
"B" is wrong, since the Precautions of OP 3337 do not require checking the CTMT boundary. "B" is plausible, since the system processes vents that come from Containment.
"C" is correct, since the Process Gas System is expected to contain high concentrations of hydrogen due to hydrogenated water in systems being serviced by the Gaseous Waste System. When this portion of the system is exposed to air, the potential exists for creating flammable concentrations of hydrogen and oxygen. Purging the piping with nitrogen inerts the gas prior to exposing it to air.
"D" is wrong, but plausible, since this is a misapplication of the precaution requiring either a Fan to be running or the stack discharge valve closed to prevent backflow from the Site Stack. Also, venting a system that is to be opened is a good tagging practice.

Technical Reference(s): OP 3337 (Rev. 19), Precautions 3.1 and 3.3
(Attach if not previously provided, including version/revision number.) OP 3337 (Rev. 19), Section 1.2, bottom of page 5
P&ID 109A (Rev. 22)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the GWS system, including the basis for each
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.5, 41.13
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 63	Tier #	2	2
K/A Statement: Ability to monitor automatic operation	Group #	2	2
of the reactor coolant system, including	K/A #	002.A3.3	
pressure, temperature, and flows	Importance Rating	4.4	4.6

Proposed Question:

Reactor power is initially at 50% with Rod Control in MANUAL.

A partial loss of feedwater heating occurs, and the RO reports RCS temperatures as follows:

- RCS Loop Thot indications have increased.
- RCS Loop Tcold indications have decreased.
- Tave indications are stable at their original value.

Assuming no operator actions have been taken, what automatically changes as a direct result of this change in RCS temperature?

a) Rod Insertion Limit setpoint increases
b) Rod Control Tref increases
c) The AMSAC System arms
d) Charging flow into the RCS decreases

Proposed Answer: A

Explanation (Optional):
"A" is correct, since the RIL setpoint increases as reactor power increases, the circuit uses RCS auctioneered high ΔT as the indication of reactor power, and ΔT has increased. RIL setpoint increases due to ΔT increasing due to the efficiency issue.
"B" is wrong, since Rod Control Tref uses selected Turbine Impulse Pressure (3MSS*PT505/506) as its input. "B" is plausible, since Tref increases as power increases (based on Turbine Impulse Pressure), and reactor power and ΔT have increased.
"C" is wrong, since AMSAC arms/disarms above or below 40% power, based on C-20, which receives input from Turbine Impulse Pressure Instruments 3MSS*PT505 and 506. "C" is plausible, since there are several inputs AMSAC, including a power input, arms on increasing power, and Reactor Power and RCS ΔT have increased on this loss of efficiency event. Also, initial reactor power is significantly lower than 100% power.
"D" is wrong, since Pzr program level increases as reactor power increases, but the circuit uses RCS Tave as the indication of reactor power, and Tave has remained stable. Pzr level setpoint hasn't changed since Tave is stable, so the Charging line flow control valve position hasn't changed.
"D" is plausible, since Pzr program level setpoint increases as reactor power increases, and power is determined by the circuit using RCS temperature inputs, and RCS ΔT, Thot, and Tcold have changed.

Technical Reference(s): OP 3350 (Rev. 7), Attachment 3
(Attach if not previously provided, Functional sheet 9 (Rev. H)
including version/revision number.) Functional sheet 11 (Rev. H)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the operation of the Reactor Coolant System under normal, abnormal, and emergency operating conditions.

Question Source: Modified Bank #404527 (Parent question attached)

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:
This question is considered “modified”, since the initiating condition has changed from an instrument failure that causes Tave to increase to a plant efficiency problem where Tave remained stable. Also, one of the distactors was changed from a change in the OTΔT trip setpoint to a change in Rod Control Tref, and another was changed from Steam Dumps arming to AMSAC arming.

Original Bank Question

Reactor Power is 80%.

RCS Loop 1 narrow range Thot RTD (3RCS*TE411A) fails high.

What is an expected plant response as a direct result of this failure?

a) Rod Insertion Limit setpoint increases.
b) Overtemperature/DeltaT trip setpoint increases.
c) Steam dump control system will ARM.
d) Charging flow will decrease.

Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Question # 64	Tier #	2	2
K/A Statement: Physical connections and/or	Group #	2	2
cause/effect relationship between the SBO diesel	K/A #	Site priority SBO.K1.1	
SBO diesel support systems	Importance Rating	3.0	3.0

Proposed Question:

The SBO Diesel is running.

Complete the following statements concerning the SBO Diesel cooling water system.

Two___(1)___ pumps supply cooling water to cool the SBO Diesel components. Heat is then transferred to a___(2)___ heat exchanger.

	(1)	(2)
a)	engine-driven	fan-cooled water-to-air
b)	engine-driven	Service Water-cooled
c)	motor-driven	fan-cooled water-to-air
d)	motor-driven	Service Water-cooled

Proposed Answer: A

Explanation (Optional):
This question is testing the applicant's knowledge of the differences between the SBO Diesel design and the Emergency Diesel design. This is significant when determining the support systems required to operate the diesel during a loss of all AC event.
"A" is correct, since the cooling water pumps are engine driven, and the heat exchanger is air cooled.
"B" is wrong, since the heat exchanger is air-cooled. "B" is plausible, since the Emergency Diesel Generators have Service Water cooled heat exchangers.
"C" and "D" are wrong, since the cooling water pumps are engine-driven pumps.
"C" and "D" are plausible, since the Emergency Diesel Generators have motor-driven cooling water pumps.

Technical Reference(s): P&ID 158B (Rev. 7)
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the function and location of the following SBO System components… Cooling Water System 1. Engine Driven Pumps 2. Temperature Regulator Valve 3. Radiator

Question Source: Bank #405008

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.41.8

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 65	Tier #	2	2
K/A Statement: Fire protection system design feature	Group #	2	2
or interlock which provides for CO2	K/A #	086.K4.6	
Proposed Question:	Importance Rating	3.0	3.3

With CO_2 initially aligned for automatic operation to the EDG fuel Oil Vaults, the following sequence of events occurs:

1. A fire breaks out in the West Diesel Fuel Oil Vault.
2. The crew enters EOP 3509, *Fire Emergency*.
3. CO_2 has NOT automatically actuated.
4. Per EOP 3509, the US directs a PEO to disable CO_2 by closing the West Diesel Fuel Oil Vault CO_2 lockout ball valve.
5. Later in the event, the fire is still burning out of control.
6. The Fire Brigade Lead recommends discharging CO_2 into the West Fuel Oil Vault.

Per OP 3341C, *Fire Protection CO_2 System*, what actions will the US direct the PEO to take, and how will the discharge timers respond to this manual discharge?

a) At the electro-mechanical pull station, the PEO will pull the locking pin and rotate the associated handle to “OPEN”. The extended discharge timer will not function.
b) At the electro-mechanical pull station, the PEO will pull the locking pin and rotate the associated handle to “OPEN”. All timers (pre-discharge, initial and extended discharge) will function as required.
c) The PEO will open the lockout ball valve, and then at the local control station the PEO will rotate the three-position key switch to “DISCHARGE”. The extended discharge timer will not function.
d) The PEO will open the lockout ball valve, and then at the local control station the PEO will rotate the three-position key switch to “DISCHARGE”. All timers (pre-discharge, initial and extended discharge) will function as required.

Proposed Answer: D

Explanation (Optional):
The US will direct the PEO to open the lockout ball valve to restore operating gas to the system. Then, and the local control station the PEO will rotate the three-position key switch to "DISCHARGE", to initiate the discharge sequence by admitting operating gas to into the system. This method will allow operating gas to fill the pre-discharge timer bottle, and then open the initial and extended discharge paths. The initial discharge timer also is driven by operating gas, so it functions to shutoff the initial discharge path after about 3 minutes. Power is still available, so the electric extended discharge timer also functions to terminate the discharge after the required time.
“A” and “B” are wrong, since the preferred method to maintain timer function is to use the three position key-lock switch. "A" and "B" are plausible, since the electro-mechanical pull station would be used if power were not available.
"C" is wrong, and "D" correct, since when using the key-lock switch, all timers function as designed. "C" is plausible, since operating the electro-mechanical pull station bypasses the extended discharge timer, and this would require the PEO to manually terminate the discharge at the appropriate time.

Technical Reference(s):	EOP 3509 (Rev. 28), steps 3.j, 3.K, and 12
(Attach if not previously provided,	OP 3341C (Rev. 18), Section 1.2, pages 4 and 5
including version/revision number.)	OP 3341C (Rev. 18), Section 4.6
	P&IDs 146A (Rev. 29) and 146E (Rev. 3)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the function and location of the following CO2 Fire Protection (FPL) system components… CO2 Pre-Discharge Timer… CO2 Initial Discharge Timer.. CO2 Extended Discharge Timer… CO2 system Lockout Valves…CO2 system Discharge Pull Stations (Abort/Discharge Switches)… CO2 system Electro-Mechanical Pull Box

Question Source: Bank #403385

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 66	Tier #	3	3
K/A Statement:	Group #	1	1
Knowledge of conservative decision making practices	K/A #	GEN.2.1.39	
Proposed Question:	Importance Rating	3.6	4.3

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

1. A storm begins impacting the intake structure at MP3.
2. The "A" Circulating Water Pump trips and the crew enters AOP 3559, *Loss of Condenser Vacuum*.
3. With condenser backpressure at 4.0 inches Hg Absolute, the crew enters AOP 3575, *Rapid Downpower* to reduce power at 5%/minute.
4. Prior to reducing turbine load, 'A' condenser backpressure peaks at 7.1 inches Hg Absolute.
5. During the downpower, rods insert below RIL.
6. The crew responds to the RIL while remaining in AOP 3575, *Rapid Downpower*.
7. When power reaches 48%, the STA reports that Tave has lowered below the minimum temperature for criticality.
8. The US directs the RO to pull rods continuously to raise temperature to within the program band.
9. The "B" Circulating Water Pump trips.
10. The crew trips the reactor and enters E-0, *Reactor Trip or Safety Injection*.

What improper action did the crew take during this event?

a) The crew was required to trip the reactor when the condenser backpressure exceeded 7.0.
b) The crew was required to enter AOP 3566, *Immediate Boration* when rods inserted below RIL.
c) The crew should not have continuously withdrawn control rods to restore RCS temperature.
d) The crew should have tripped the turbine and entered AOP 3550, *Turbine Trip* when the "B" Circ Water Pump tripped.

Proposed Answer: C

Explanation (Optional): This event is based on the Salem marsh grass event in SOER 94-1 Non-Conservative Decisions, where operators inappropriately pulled rods continuously with an unstable secondary plant, resulting in a safety injection. "A" is wrong, since AOP 3559 does not require a reactor trip unless condenser backpressure reaches 7.5 inches Hg Absolute. "A" is plausible, since condenser pressure is elevated, and a reactor trip requirement and the auto turbine trip occurs at 7.6 inches Hg Absolute.
"B" is wrong, since AOP 3575 provides adequate guidance for immediate boration with rods below RIL. "B" is plausible, since RIL is normally an entry condition for AOP 3566.
"C" is correct, since adding positive reactivity is never an appropriate way to address unstable plant conditions. It is non-conservative to withdraw control rods in an attempt to restore primary coolant temperature during a transient.
"D" is wrong, since, AOP 3559 step 4 directs a reactor trip when both circulating water pumps are tripped in one condenser. "D" is plausible, since power is below P-9 and a turbine trip would not directly cause a reactor trip.

Technical Reference(s): AOP 3575 (Rev. 27), step 7 (pages 16-17)
(Attach if not previously provided, including version/revision number.) AOP 3559 (Rev. 13), step 4 (page 6) & Foldout Page
AOP 3566 (Rev. 14), Entry Condition (page 2)
OP-AP-300 (Rev. 22), Att. 2, page 4

Proposed references to be provided to applicants during examination: None

Learning Objective: Implement the reactivity standards when performing any plant manipulation that has the potential to change reactor power

Question Source: Modified Bank# 409102 (Parent question attached)

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10, 43.5

Comments:

This question is considered "modified" since the power level was changed in the stem and a new abnormal Condenser backpressure condition was added to the stem. Also, new distractor "A" was added, related to Condenser pressure.

Previous question:

During a storm, screen DP starts increasing on the "A" Circulating Water Pump; and the following sequence of events occurs:

1. The "A" Circulating Water Pump trips.
2. The crew enters AOP 3575 *Rapid Downpower* and starts reducing power at 5%/minute.
3. During the downpower, rods insert below RIL due to Tave/Tref error.
4. The crew commences rapid boration per AOP 3575 *Rapid Downpower*.
5. When power reaches 28%, the STA notices that Tave has lowered below the minimum temperature for criticality.
6. The US directs the RO to pull rods continuously to raise temperature to within the program band.
7. The "B" Circulating Water Pump trips.
8. Condenser backpressure in the "A" condenser bay increases to 6 inches Hg absolute.
9. The crew trips the reactor and enters E-0 *Reactor Trip or Safety Injection*.

What improper action did the US take during this event?

a) The US was required to direct a reactor trip when the "A" Circulating Water Pump tripped.
b) The US was required to enter AOP 3566 *Immediate Boration* when rods inserted below RIL.
c) The US should not have directed the RO to continuously withdraw control rods to restore RCS temperature.
d) The US should have directed a turbine trip and entered AOP 3550 *Turbine Trip*, rather than trip the reactor.

Answer: C

Examination Outline Cross-reference:	Level	RO	SRO
Question # 67	Tier #	3	3
K/A Statement: Ability to use procedures to determine the effects on reactivity of plant changes such as RCS temperature, secondary plant, fuel depletion, etc.	Group #	1	1
	K/A #	GEN.2.1.43	
	Importance Rating	4.1	4.3

Proposed Question:

Initial conditions:

- The plant is operating at 100% power and stable.
- The calorimetric is selected to Steam Flow
- The calorimetric 10-minute average power indicates 3650.0 MWth.

Which of the following would **NOT** require a downpower in accordance with OP-AP-300, *Reactivity Management*?

a) Raising Letdown temperature
b) Increasing Auxiliary Steam System supply flow
c) Removing a 1st Point Feedwater Heater from service
d) Increasing Blowdown flow

Proposed Answer: A

Explanation (Optional):
OP-AP-300, *Reactivity Management* states: "**ENSURE** power is reduced to maintain margin to less than 100.0 percent before conducting evolutions that are known to add positive reactivity." Examples are provided which include: lowering letdown temperature, increasing blowdown flow rate, increasing aux steam supply flow, and changing feed water conditions.
"A" is correct as raising letdown temperature will cause the letdown demineralizers to release boron, which adds negative reactivity to the core.
"B", "C", and "D" are wrong, since each of these evolutions would cause a positive reactivity addition and would require a power reduction before starting the evolution.
"B" is plausible since increasing Aux Steam flow would cause an indirect positive reactivity addition, since it would increase Main Steam flow. Also, Aux Steam can be supplied by the aux boilers during plant startups, and if the aux boilers were supplying the Aux Steam header, a power reduction would not be necessary.
"C" is plausible as removing a first water feed heater from service causes an indirect positive reactivity addition due to the resultant colder feedwater.
"D" is plausible, since increasing blowdown flow indirectly causes reactor power to increase due to causing increased feed flow into the SGs.

Technical Reference(s): OP-AP-300 (Rev.22), Attachment 2, page 1
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None.
Learning Objective: Implement the reactivity standards when performing any plant manipulation that has the potential to change reactor power
Question Source: Bank #409085
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or fundamental knowledge
10 CFR Part 55 Content: 55.41.1, 41.6, 41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 68	Tier #	3	3
K/A Statement:	Group #	2	2
Knowledge of the process for managing maintenance activities	K/A #	GEN.2.2.17	
during power operations such as risk assessments, work prioritization, and coordination with the transmission system	Importance Rating	2.6	3.8

Proposed Question:

Due to severe transmission system problems, Millstone Station receives the following notification:

"This is ISO New England, we are in a transmission system emergency. Reduce net combined Millstone Station output to 1600 MW."

Per C OP 200.8, *Response to ISO New England/CONVEX Notifications and Alerts*, Millstone Power Station is required to achieve this load reduction within how many minutes of the notification?

a) 15 minutes
b) 20 minutes
c) 25 minutes
d) 30 minutes

Proposed Answer: C

Explanation (Optional):
"C" is correct, and "A", "B", and "D" are wrong, since COP 200.8 provides instructions to achieve an emergency reduction of net combined Millstone Station electrical generation within 25 minutes of notification when electrical distribution system conditions warrant. "A", "B", and "D" are plausible, since each distractor is arranged in 5 minute increments, and all are much more rapid than a normal downpower rate.

Technical Reference(s): C OP 200.8 (Rev.7), Section 1.2 (page 2)
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the actions which must be completed within 25 minutes of receiving a Convex request for emergency load reduction
Question Source: Bank #369787
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or fundamental knowledge
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 69	Tier #	3	3
K/A Statement:	Group #	2	2
Knowledge of the process for managing maintenance activities during shutdown operations, such as risk assessments, work prioritization, etc.	K/A #	GEN.2.2.18	
	Importance Rating	2.6	3.9

Proposed Question:

The plant is in MODE 6, with a refueling outage in progress.

Which ONE of the following is one of the five "Key Safety Functions" specified in OU-M3-201, *Shutdown Safety Assessment Checklist*?

a) Heat Sink
b) Power Availability
c) AC/DC Instrumentation
d) Cavity Level

Proposed Answer: B

Explanation (Optional):
Per OU-M3-201, the five key safety functions when shutdown are:
Decay Heat Removal, Inventory, Power Availability, and Containment
"A" is wrong, since Heat Sink is not a key safety function. "A" is plausible because Heat Sink is one of the CSF Status Trees.
"B" is correct, since Power Availability is a key safety function.
"C" is wrong, since AC/ DC Instrumentation are not key safety functions. "C" is plausible, since AC/DC Instrumentation are checked per OU-M3-201, and are listed as supporting "defense in depth."
"D" is wrong, since Cavity Level is not a key safety function. "D" is plausible, since the Cavity Level is a sub-category checked in the Decay Heat Removal category.

Technical Reference(s): OU-M3-201 (Rev.26), Section 3.1
(Attach if not previously provided, including version/revision number.) OU-M3-201 (Rev.26), Attachment 1, page 2
OU-M3-201 (Rev.26), Attachment 7, page 2
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss the general guidance for shutdown nuclear safety specified in the shutdown risk management procedure (294417)
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or fundamental knowledge
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 70	Tier #	3	3
K/A Statement: Knowledge of criteria or conditions	Group #	1	1
that require plant-wide announcements, such as	K/A #	GEN.2.1.14	
pump starts, reactor trips, mode changes, etc.	Importance Rating	3.1	3.1

Proposed Question:

Answer the following two questions regarding plant announcement standards required per OP-AA-100, *Conduct of Operations*.

(1) What are the MINIMUM size motors which require a plant announcement prior to a planned start or stop of the motor?
(2) What is this announcement required to direct plant personnel to stand clear of?

a) (1) Large 480 volt or greater
(2) The equipment only
b) (1) Large 480 volt or greater
(2) The equipment and its associated electrical switchgear
c) (1) 4kv or greater
(2) The equipment only
d) (1) 4kv or greater
(2) The equipment and its associated electrical switchgear

Proposed Answer: B

Explanation (Optional):
OP-AA-100 requires plant announcements prior to planned starting or stopping of certain motors.
"A" is wrong, since the announcement is required to direct plant personnel to stand clear of the equipment and its associated electrical switchgear. "A" is plausible, since the equipment is a subset of the correct answer, and is where the noise and rotation will occur.
"B" is correct, since the motor size that requires an announcement are large 480 volt or greater, and the announcement will direct plant personnel to stand clear of the equipment and its associated electrical switchgear.
"C" and "D" are wrong, since the minimum size motors that require an announcement are large 480 volt or greater. "C" and "D" are plausible as the next highest voltage motor (from 480 volt) is 4KV motors.

Technical Reference(s): OP-AA-100 (Rev. 38), Section 16
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: List the information that will be announced via the paging system.
Question Source: Robinson 2013 NRC Exam
Question History: Last NRC Exam Robinson 2013 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 71	Tier #	3	3
K/A Statement:	Group #		
Knowledge of radiological safety procedures pertaining to	K/A #	GEN.2.3.13	
Licensed operator duties, such as response to rad monitor alarms, CTMT entry requirements, fuel handling, access to locked hi rad areas, aligning filters, etc.	Importance rating	3.4	3.8

Proposed Question:

The plant is in MODE 5 with preparations being made for the initial Containment entry, and current conditions are as follows:

- An operator is required to enter the MIDS area inside Containment.
- The area is posted as a locked high radiation area.

What requirements exist for this entry in addition to the normal requirements for entering a locked high radiation area?

a) The Incore System Drives must be tagged out.
b) An access control guard must be stationed at the locked entrance.
c) A backup electronic alarming dosimeter is required to be worn.
d) Authorization for key usage must be obtained by Radiation Protection (RP) Supervision.

Proposed Answer: A

Explanation (Optional):
"A" is correct, since the Incore Drives must be tagged out.
"B" is wrong, but plausible, since this is a requirement for both a locked high radiation and the MIDS area when work requires the lock to be defeated.
"C" is wrong, since there is no requirement for a backup dosimeter. "C" is plausible, since potentially lethal radiation dose rates may exist in the area.
"D" is wrong, but plausible, since this is required for all locked high radiation areas.

Technical Reference(s): RP-AA-201 (Rev. 10), pages 6-8 and 11-13.
(Attach if not previously provided, OP 3361A (Rev. 11), Section 4.1
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: Describe Management responsibilities with regard to MIDS area entries (251954)
Question Source: Bank #369988
Question History: Last NRC Exam MP3 2011 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.12, 43.4
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 72	Tier #	3	3
K/A Statement:	Group #	3	3
Knowledge of radiation exposure limits under	K/A #	GEN.2.3.4	
normal or emergency conditions	Importance Rating	3.2	3.7

Proposed Question:

Complete the following statement concerning annual TEDE dose limits to qualified radiation workers.

The 10CFR20 limit is___(1)____, and the Dominion administrative limit for the home Dominion site is___(2)____ without special authorization.

	(1)	(2)
a)	3,000 mrem	1,000 mrem
b)	3,000 mrem	2,000 mrem
c)	5,000 mrem	1,000 mrem
d)	5,000 mrem	2,000 mrem

Proposed Answer: D

Explanation (Optional):
"A" and "B" are wrong, since the 10CFR20 limit is 5,000 mrem per year.
“A” and “B” are plausible, since the listed 10CFR20 limit is above the admin limit, and within the actual 10CFR limit.
"C" is wrong, and "D" correct, since the Dominion administrative limit is 2,000 mrem per year without special authorization. “C” is plausible, since the lower admin limit would be conservative, and it is above the amount operators routinely receive during a calendar year.

Technical Reference(s): RP-AA-105 (Rev. 2), Attachment 1, page 1
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the personnel radiation exposure limits under normal and emergency conditions (251946).
Question Source: Bank #369974
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.12, 43.4
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 73	Tier #	3	3
K/A Statement:	Group #	4	4
Knowledge of abnormal condition procedures	K/A #	GEN.2.4.11	
Proposed Question:	Importance Rating	4.0	4.2

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

- Reactor Power is 8%.
- No RPS bistables are in the tripped condition.
- All Control Systems are selected to Channel 1 for plant control.

The following sequence of events occur:

1. Numerous Main Board annunciators illuminate.
2. The BOP operator reports VIAC 2 is deenergized.

Answer the following two questions.

(1) Did the reactor immediately trip due to the loss of VIAC 2?
(2) If the reactor did not trip, which AOP is the crew required to enter first? If the reactor did trip, how if at all, will AOP 3564, *Loss of One Protective System Channel* be implemented while in the EOP network?

a) (1) The reactor did NOT trip.
(2) The crew will enter AOP 3564, *Loss of one Protective System Channel*.
b) (1) The reactor did NOT trip.
(2) The crew will enter AOP 3581, *Immediate Operator Actions*.
c) (1) The reactor DID trip.
(2) While in ES-0.1, *Reactor Trip Response*, AOP 3564 steps may be implemented, as necessary, to gain control of impacted systems.
d) (1) The reactor DID trip.
(2) AOP 3564 may NOT be implemented until ES-0.1, *Reactor Trip Response*, is completed, since EOPs are prioritized above AOPs.

Proposed Answer: C

Explanation (Optional):
The Intermediate Range (IR) NI's are powered from protective channel I and II. If VIAC-2 deenergizes, IR 36 will de-energize and the associated bistable High Flux Reactor trip will go to the trip condition. This is a 1 of 2 coincidence, so the reactor trips. The procedure flowpath will be E-0 to ES-0.1. While in ES-0.1, there will be mitigative steps in AOP 3564 that can be used in parallel. The EOP User's guide (OP 3272 page 21) states that subordinate AOP's may be used, as necessary, on a step by step basis while carrying out EOP's.
"A" and "B" are wrong, since the reactor trips on loss of power to Intermediate Range Channel N36. "A" and "B" are plausible since the reactor would not trip above 10% power (P10), where the IRNI reactor trip is blocked. Also, with the plant below 30% power, AOP 3581 is not applicable, so the crew would enter AOP 3564. Below 30%, the crew would enter AOP 3581 for any applicable control system malfunctions.
"C" is correct, and "D" wrong, since AOPs are allowed to be used in parallel with EOPs to address specific problems related to the AOP, as long as the EOP receives priority. ES-0.1 use alone is wrong in this case as it was not written assuming instrument channels are available. "D" is plausible since EOP implementation is prioritized above AOPs according to general rules of usage.

Technical Reference(s): AOP 3581 (Rev. 5), Section 2.0
(Attach if not previously provided, including version/revision number.) OP 3272 (Rev.10), Section 1.8, page 21
Functional Sheet 4 (Rev. G)
EE-1BG (Rev. 30)

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the usage of AOP's while in EOP's

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 74	Tier #	3	3
K/A Statement:	Group #	4	4
Knowledge of annunciator alarms, indications, or	K/A #	GEN.2.4.31	
response procedures	Importance Rating	4.2	4.1

Proposed Question:

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

1. The RPCCW HX SW FLOW HI/LO (MB1C, 1-1A) annunciator is received.
2. The RO reports the following Service Water flow indications on MB1:

- 3SWP-FI43A, "SERVICE WTR DIS FLOW" "RPCCW HX" : 11,800 gpm
- 3SWP-FI43B, "SERVICE WTR DIS FLOW" "RPCCW HX" : 8,800 gpm

What is a potential cause of these indications; and what procedure flowpath will best mitigate the event?

a) A pipe break exists on the "A" Train. The crew will enter AOP 3560, *Loss of Service Water*, since AOP 3560 will provide direction to isolate the break.
b) A pipe break exists on the "A" Train. The crew will enter the ARP to isolate the break, and then go to AOP 3560, *Loss of Service Water*.
c) Heat Exchanger fouling exists on the "B" Train. The crew will enter AOP 3560, *Loss of Service Water*, since AOP 3560 will direct the crew to place the standby RPCCW Heat Exchanger in service.
d) Heat Exchanger fouling exists on the "B" Train. The crew will enter the ARP to place the standby RPCCW Heat Exchanger in service, and then go to AOP 3560, *Loss of Service Water*.

Proposed Answer: B

Explanation (Optional):
Normal SWP Flowrate to an RPCCW HX is about 8,800 gpm, so Train "A" is showing excessive flow, indicating a potential pipe break, while train "B" is showing a normal flowrate ("C" and "D" wrong). "C" and "D" are plausible, since flow is lower in the "B" train than the "A" train, AOP 3560 takes actions for a loss of service water, and the ARP directs the actions in "D" if flow is low. "B" is correct, since a leak downstream of 3SWP*MOV50A is isolable, and will be isolated by the ARP. "A" is wrong, since AOP 3560 does not address the initial required actions for a pipe break, since the initial actions are provided in the ARPs. "A" is plausible, since AOP 3560 takes actions for a loss of service water.

Technical Reference(s): AOP 3560 (Rev. 12), Note prior to step 1
(Attach if not previously provided, OP 3353.MB1C (Rev. 15), 1-1A, Setpoint and step 8.2
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: Given a set of plant conditions, properly apply the notes and cautions of AOP 3560
Question Source: Bank #406927
Question History: Last NRC Exam Millstone 3 2009 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 75	Tier #	3	3
K/A Statement:	Group #	4	4
Ability to verify that alarms are consistent with plant conditions	K/A #	GEN.2.4.46	
	Importance Rating	4.2	4.2

Proposed Question:

The unit was at 100% power when the reactor trips due to a turbine trip.

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

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

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

a) The TDFP TRIP annunciators should not be lit, since no signal trips them if SIS does not occur. This may have been the cause of the Turbine trip.
b) The DWST LEVEL LO annunciator should not be lit, since the AFW recirculation path is always aligned to the DWST. Aux Feed Pump suctions are required to be swapped to the CST.
c) The FEEDWATER HEATER LEVEL LO annunciators should not be lit, since the Main Feed Pump recirc valves should have opened on the Feedwater Isolation. Operators are required to trip the running 4th Point Heater Drain Pumps.
d) The MOIST SEP WATER LEVEL HI annunciators should not be lit, since the Moisture Separator Drain Tank Emergency Drain Valves should have opened. This may have been the cause of the turbine trip.

Proposed Answer: D

Explanation (Optional):

"A" is wrong, since the TDMFPs do receive a trip signal after a time delay indirectly from P-4, preventing feed header over-pressurization on a trip. "A" is plausible since the trip is indirect, after a time delay, and SIS immediately trips the TDAFW Pumps.
"B" is wrong, since AFW takes a suction on the DWST, lowering DWST level. "B" is plausible, since the AFW swap-over to the CST is required if DWST LO-LO is received, and the AFW pumps do recirc back to the DWST. Also, this annunciator is not be expected during normal plant operations.
"C" is wrong, since the source of extraction steam to the heaters are the HP and LP turbines, and these have been isolated on the turbine trip. Heater levels drop and low level alarms are expected. "C" is plausible, since the heater strings remain unisolated on a reactor trip.
"D" is correct, since high level in the MSR Drain Tank sends a signal to open the emergency drain valve. If level continues to increase, the MOIST SEP WATER LEVEL HI results in a turbine trip. This protects the turbine from water entering from the Moisture Separator.

Technical Reference(s): (Attach if not previously provided, including version/revision number.)	OP 3353.MB5A (Rev. 12), 5-5, Corrective Action 2.6
	OP 3353.MB6A (Rev. 13), 5-8
	OP 3353.MB6A (Rev. 13), 1-1B, Corrective Actions 1 and 3
	OP 3353.MB6A (Rev. 13), 3-4
	OP 3353.MB7B (Rev. 3), 5-1
	P&ID 125A (Rev. 35)

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #403912

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.5, 41.10, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 76	Tier #		1
K/A Statement:	Group #		1
Ability to interpret and execute procedure steps during a Pzr vapor space accident	K/A #	APE.008.GEN.2.1.20	
	Importance Rating		4.6

Proposed Question:

Initial conditions:

- The plant is at 100% power.
- The "A" PORV Block Valve is red tagged in the deenergized, CLOSED position.

SIS actuates due to a leaking Pzr Safety Valve, resulting in the following sequence of events:

1. The crew enters E-1, *Loss or Reactor or Secondary Coolant*.
2. The Pzr is empty.
3. RCS pressure is 1300 psia and slowly decreasing.
4. SG pressures are stable at 1100 psig.

The crew is performing E-1, step 8, "Check Stopping RHR Pumps", when the following sequence of events occurs:

1. The RO reports the "B" PORV indicates OPEN.
2. Prior to US direction, the RO attempts to close the "B" PORV, and it does NOT close.
3. Prior to US direction, the RO closes the "B" PORV Block Valve.
4. The RO reports RCS pressure is 1250 psia and slowly decreasing.

Complete the following statement concerning the action the US is required to direct with the "B" PORV Block Valve, and the procedure the US will be required to transition to when exiting E-1.

Based on the current condition with both PORV Block Valves closed, E-1 will direct the crew to__(1)__, and the US will go to__(2)__when transitioning from E-1.

	(1)	(2)
a)	leave the "B" PORV Block Valve closed	ES-1.1, *SI Termination*
b)	leave the "B" PORV Block Valve closed	ES-1.2, *Post LOCA Cooldown and Depressurization*
c)	reopen the "B" PORV Block Valve	ES-1.2, *Post LOCA Cooldown and Depressurization*
d)	reopen the "B" PORV Block Valve	ES-1.3, *Transfer to Cold Leg Recirculation*

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions, and interpret a procedure step that would normally require the crew to re-open a Block Valve. The question also requires the applicant to determine which ES procedure the crew is required to transition to. Based on the CAUTION prior to E-1, step 5, the action to close a stuck open PORV is a continuous action step. It requires the crew to close the stuck open PORV, and if it fails to close, close its Block Valve, so the RO's actions were correct.

"A" is wrong, since after the Block Valve is closed, conditions are restored to the conditions that existed before the PORV failed open. RCS pressure was slowly decreasing, so when the crew reaches E-1, step 9, with SG pressure stable and RCS pressure decreasing, the crew will proceed to E-1, step 12, which will direct the crew to ES-1.2 with RCS pressure above 300 psia.

"A" is plausible, since with either SG pressures decreasing or RCS pressure increasing, E-1, step 9 would loop the crew back to E-1, step 1, and at Step 6, the crew would assess plant conditions for a transition to ES-1.1.

"B" is correct, since the crew is required to keep the "B" Block Valve closed, since one Block Valve is not available, and the other is being used to isolate a stuck open PORV. Also, when the crew reaches E-1, step 9, with SG pressure stable and RCS pressure decreasing, the crew will proceed to E-1, step 12, which will direct the crew to ES-1.2 with RCS pressure above 300 psia.

"C" and "D" are wrong, since the crew is required to keep the "B" Block Valve closed, since one Block Valve is not available, and the other is being used to isolate a stuck open PORV.

"C" and "D" are plausible, since E-1, step 5.b has the crew check at least one Block Valve open, which normally is desired, and if neither are open, the crew would normally open at least one block valve. But the RNO has the crew keep the block valves closed if they are unavailable or are being used to isolate a stuck open PORV. Also, if a Block Valve were reopened, RCS pressure would continue to decrease, and if pressure dropped below 300 psia with RHR Pump flow, the crew would proceed to E-1, step 14, where the crew would wait for RWST Lo-Lo level conditions and transition to ES-1.3.

Technical Reference(s): E-1 (Rev. 27), steps 5-12
(Attach if not previously provided, including version/revision number.)

Proposed references to be provided to applicants during examination: None

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

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55. 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 77	Tier #		1
K/A Statement: Ability to determine and interpret actions to be taken if limits for PTS are violated during a large break LOCA	Group #		1
	K/A #	EPE.011.EA2.14	
	Importance Rating		4.0

Proposed Question:

A large-break LOCA has occurred, and current conditions are as follows:

- The crew has entered ES-1.3, *Transfer to Cold Leg Recirculation*.
- The operators are aligning the RCS for cold leg recirculation per ES-1.3, step 3.

The STA informs the SM that a red path just came in on the Integrity CSF Status Tree.

How is the US required to address the red path on RCS Integrity?

a) Transition to FR-P.1, *Response to Imminent Pressurized Thermal Shock*.
b) Perform FR-P.1 and ES-1.3 in parallel, with FR-P.1 steps receiving priority. The steps of ES-1.3 needed to ensure success of FR-P.1 are performed in parallel.
c) Continue in ES-1.3 until the cold leg recirculation alignment is completed, and then transition to FR-P.1, *Response to Imminent Pressurized Thermal Shock*.
d) Complete ES-1.3, and then transition to FR-P.1, *Response to Imminent Pressurized Thermal Shock*.

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions during an emergency, and determine to proceed in the current event-specific emergency sub-procedure, even though general rules of status tree usage would require a transition to a functional restoration procedure.
"A", "B", and "D" are wrong, and "C" correct, since completing the alignment for Cold Leg Recirculation is the highest priority action when RWST Lo-Lo level is reached to ensure long term cooling. Operators will be directed in ES-1.3 and FR-P.1 to complete the Cold Leg Recirculation lineup prior to addressing the PTS condition.
"A" is plausible, since normally in the EOPs, Status Tree colors of Orange or Red require immediate transition from the procedure in effect to the appropriate FR Procedure.
"B" is plausible, since this gives the FR Procedure priority, and the general EOP rules of use allow for parallel procedure use.
"D" is plausible, since the Cold Leg Recirculation portion of ES-1.3 receives priority over FR-P.1.

Technical Reference(s): OP 3272 (Rev. 10), Attachment 4, sheets 2 and 4
(Attach if not previously provided, including version/revision number.) ES-1.3 (Rev. 18), steps 1-7, including notes prior to steps 1 and 7.

Proposed references to be provided to applicants during examination: None

Learning Objective: Discuss conditions which require transition to other procedures from ES - 1.3 and ES - 1.4

Question Source: Bank #407934

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 78	Tier #		1
K/A Statement: Ability to determine and interpret	Group #		1
Proper amperage of running LPI/RHR pump on a loss	K/A #	APE025.AA2.1	
of residual heat removal system	Importance Rating		2.9

Proposed Question:

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

- The "A" RHR Pump: Running in the "Cooldown" Mode
- RCS temperature: 150°F
- RCS pressure: 150 psia
- The Pressurizer: Solid
- RCPs: None running
- The "A" Charging Pump: Running
- All Steam Generators: 50% Narrow Range
- "B" Electrical Train: Train outage is in progress, NOT available for restoration

The following sequence of events occurs:

1. The RO reports zero amps are indicated for the "A" RHR Pump.
2. The crew enters EOP 3505, *Loss of Shutdown Cooling and/or RCS Inventory*.
3. The RO reports RCS temperature has started increasing.

What actions are required to be directed by the US per EOP 3505 to remove decay heat?

a) Open both PORVs, and feed the RCS using one Charging Pump from the RWST.
b) Open both PORVs, and feed the RCS using one SI Pump from the RWST.
c) Throttle open the Charging Flow Control Valve to raise RCS pressure to >170 psia, and open the SG Atmospheric Relief Valves.
d) Start one Reactor Coolant Pump, check proper differential pressure across its seals, and open the SG Atmospheric Relief Valves.

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions, and select the appropriate Attachment in EOP 3505, and determine the actions to be taken per that attachment.
"A" and "B" are wrong, but plausible, since bleed and feed is would be established only if natural circulation cooling is unsuccessful. The charging pump is the preferred feed source, and the SI Pump is the backup source of feed.
C" is correct, since the RCS is already filled and steam generators are available. The procedure will direct the US to establish conditions for natural circulation, so RCS pressure is increased to >170 psia ensure subcooled natural circulation cooling will occur, and then the atmospheric relief valves are used to dump steam from the steam generators.
"D" is wrong, but plausible, since forced cooling would be established, only if a RCP is already running.

Technical Reference(s): EOP 3505 (Rev. 16), Section 2.1
(Attach if not previously provided, EOP 3505 (Rev. 16), Step 9
including version/revision number.) EOP 3505 (Rev. 16), Att. "A", steps A.2, A.5–A.8
Proposed references to be provided to applicants during examination: None
Learning
Objective: Given a set of plant conditions, determine the required actions to be taken per EOP 3505.
Question Source: Bank #407795
Question History: Last NRC Exam Millstone 3 2011 NRC
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 79	Tier #		1
K/A Statement:	Group #		1
Ability to prioritize and interpret the significance of	K/A #	APE.056.GEN.2.4.45	
each annunciator or alarm during a loss of offsite power	Importance Rating		4.3

Proposed Question:

Offsite power has been lost, and initial conditions are as follows:

- The crew is conducting a RCS cool down and depressurization per ES-0.4, *Natural Circulation Cooldown with Steam Void in Vessel (Without RVLMS)*.
- RCS Hot Leg temperatures: 390°F & stable
- RCS Pressure: 725 psia & decreasing

The following sequence of events occurs:

1. The PRESSURIZER LEVEL HI annunciator is received on MB4A.
2. The RO reports PZR level is 91% and increasing.

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

a) Stop depressurizing, and increase RCS pressure by 100 psi using Pzr heaters
b) Maintain stable Pressurizer level by reducing charging flow into the RCS
c) Decrease RCS hot leg wide range temperatures to 350°F
d) Equalize Charging and Letdown flows

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess plant conditions, and determine specific action required based on an abnormal condition, requiring detailed knowledge of the procedure which goes beyond knowing the overall procedure strategy, and which is beyond system knowledge. In ES-0.4, a cooldown rate that will cause a void in the reactor vessel is allowed in order to reach COLD SHUTDOWN faster. A stepped cooldown and depressurization is being conducted, and without RVLMS, PZR level is being used as a backup indication for reactor vessel void size. During the cooldown steps, charging flow is increased to maintain Pzr level to compensate for RCS coolant shrinkage, and during the depressurization steps, Charging and Letdown flows are equalized to ensure any changes in Pzr level are due to changes in the void size in the reactor vessel. Based on plant conditions, the crew is in ES-0.4, step 16b. Pzr level >90% indicates the void may be too large, so one RCS depressurization termination criterion in step 16b is Pzr level >90%. This is met, so the crew is required to stop the depressurization.
"A" is correct, since with Pzr level >90%, the crew is required to stop the depressurization, and then repressurize the RCS to collapse the void and cool the head with cooler RCS water as the void collapses.
"B" is wrong, since repressurizing to collapse the void and cool the head will result in Pzr level changing.
“B” is plausible, since this will be performed in the next cooldown step in ES-0.4 after the depressurization step is complete, assuming Pzr level is below 91%.
“C”" is wrong, since the cooldown is not performed during the depressurization steps. “C” is plausible, since this is the next step to be performed if Pzr level is <90%, and the depressurization step is complete. And, with a normal cooldown, while not on natural circulation, cooling down the RCS will aid in cooling the reactor vessel head.
"D" is wrong, but plausible, since in ES-0.4, Charging and Letdown flows are equalized prior to performing the depressurization steps.

Technical Reference(s): ES-0.4 (Rev. 13-1), steps 14 through 19
(Attach if not previously provided, including version/revision number.) OP 3353.MB4A (Rev. 6), 3-1

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #408705

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.10, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 80	Tier #		1
K/A Statement: Ability to determine and interpret location and isolation of leaks during a loss of instrument air	Group #		1
	K/A #	APE.065.AA2.3	
	Importance Rating		2.9

Proposed Question:

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

- RCS temperature is 300°F.
- The "A" Train of RHR is in service in the "Cooldown" Mode.
- SG levels are being maintained with the MDAFW Pumps.

The following sequence of events occurs:

1. Instrument Air (IAS) pressure starts decreasing, and the crew enters AOP 3562, *Loss of Instrument Air*.
2. The RO reports IAS pressure has decreased to zero psig as indicated on 3IAS-PI29 (MB1).
3. A PEO reports a major air leak in the main IAS header, above the west side of the "B" Condenser Bay.
4. The RO reports IAS pressure in Containment has decreased to zero psig as indicated on MB1.
5. The crew completes all required actions in OP 3562, steps 1-11, and is looping through the procedure awaiting restoration of IAS system pressure.
6. The PEO isolates the break by closing the two closest IAS header isolation valves (3IAS-V62 and V63), one on each side of the break.

The US is considering the following four actions specified in AOP 3562:

1. AOP 3562, step 13, restoring Instrument Air pressure to Containment
2. OP 3330C, *Reactor Plant Chilled Water System*, establishing CDS to Containment
3. AOP 3562, Attachment H, *Placing Train B RHR in Service as Second Train of RHR*
4. OP 3322, *Auxiliary Feedwater System*, stopping the TDAFW Pump

Now that IAS pressure has been restored to 110 psig on 3IAS-PI29, which three of the above actions are required to be performed by the crew per AOP 3562?

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

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since it requires the applicant to assess abnormal plant conditions, and select procedures and sections of procedures, and determine when to implement procedure attachments. It also requires applicants to understand which actions are taken in the body of AOP 3562, and which actions require the use of Attachments or portions of other procedures. The crew has been looping through AOP 3562, awaiting restoration of air pressure. When the PEO isolates the break, air pressure is restored to the Turbine Building and the Auxiliary Building, since the Turbine Building air header is a ring header, and the Aux Building supply header is outside of the isolated section of piping. Air has not been restored to Containment, since a fail-closed AOV is in that line. When air pressure is restored, AOP 3562, step 12 will direct the crew to proceed forward in AOP 3562.
"A" is wrong, since AOP 3562, step 13--which is not performed until after air pressure is restored--will direct the crew to check IAS aligned to Ctmt, and if needed (it is needed), open 3IAS*PV15 to restore IAS to Ctmt. "A" is plausible, since this step would not be needed if the IAS path to CTMT had not automatically isolated on the loss of air pressure, and would not be possible to perform if isolating the IAS break had isolated the IAS path to Ctmt, which is downstream of the Turbine Building.
B" is correct, since the three actions specified are required per AOP 3562 now that air pressure has been restored, and air pressure is available to each of these areas to accomplish these steps. Also, the one step it does not specify, AOP 3562, Attachment H, is not directed to be used after AOP 3562, steps 13 forward to the end of the procedure is being implemented. It would be directed in step 11.f while air pressure was lost if RCS temperature was less than 260°F. Also, step 17 will direct restoration of RHR alignment, but through use of OP 3310A, not AOP 3562, Attachment H.
"C" is wrong, since AOP 3562, step 15 will direct the crew to use OP 3330C to establish CDS to Containment. "C" is plausible, since this would not be performed if CDS were not required in CTMT in this lower MODE, and would not be possible if isolating the IAS break had isolated the IAS path to the Aux Building, which is downstream of the Turbine Building.
"D" is wrong, since AOP 3562, step 16 RNO directs the crew to use OP 3322 to stop the TDAFW pump if it is not required to maintain SG levels. "D" is plausible, since this would not be performed if AFW were not needed in this lower MODE, or if the TDAFW Pump were needed to maintain SG levels, and would not be possible if isolating the IAS break had isolated the IAS path to the ESF Building, which is downstream of the Turbine Building.

Technical Reference(s): AOP 3562 (Rev. 17), steps 11 through 17
(Attach if not previously provided, including version/revision number.) P&ID 138B (Rev. 36), and 138C (Rev. 34)

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per AOP 3562.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 81	Tier #		1
K/A Statement: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures during LOCA outside CTMT	Group #		1
	K/A #	W/E04.GEN.2.4.4	
	Importance Rating		4.7

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 30 minutes, RCS pressure increases to and stabilizes at 2350 psia, cycling with the PZR PORVs.
3. The crew transitions from E-0 to ECA-1.2, *LOCA Outside Containment*.
4. RWST level is 900,000 gallons and slowly decreasing.
5. Pressurizer level is 65% and slowly increasing.
6. While attempting to isolate the break, the final valve closed by the crew 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 observing the Real-Time trend reports that the PORVs are now cycling at a significantly faster rate.

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

a) E-0, *Reactor Trip or Safety Injection*
b) ES-1.1, *SI Termination*
c) E-1, *Loss of Reactor or Secondary Coolant*
d) ECA-1.1, *Loss of Emergency Coolant Recirculation*

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions, and make the appropriate procedure transition out of an emergency contingency action procedure. This transition requires the use of backup indications, since the primary indication, RCS pressure increasing, is not available with the RCS at the PORV setpoint.
“A” is wrong, since ECA-1.2 transitions the crew to either E-1, if the break is isolated, or ECA-1.1 if the break is not isolated. “A” is plausible, since the break is isolated, and numerous EOPs transition the crew back to “the procedure and step in effect” if the current procedure successfully mitigates the event at hand, and E-0 would then loop the crew back to its diagnostic steps.
“B” is wrong, since ECA-1.2 transitions the crew to either E-1, if the break is isolated, or ECA-1.1 if the break is not isolated. “B” is plausible, since the goal with the break isolated is to terminate SI, and with the break isolated, the crew will rapidly transition out of E-1 to ES-1.1, *SI Termination*.
“C” is correct, and “D” wrong, since, even though RCS pressure is being held stable by the PORVs, the crew is required to transition to E-1. This is because it can be determined that the break is isolated since the PORV cycling rate has changed. With the break isolated, mass loss out the break has stopped, and mass is still being injected at the same rate from High Head Safety Injection. This requires the applicant to apply the Notes prior to step 1 of ECA-1.2, which direct the crew to monitor diverse indications when RCS pressure is cycling at the PORV setpoint.
"D" is plausible since normally, when the break is isolated, RCS pressure will start increasing, and this does not occur with the RCS at the PORV setpoint. Based on RCS pressure trend in the absence of PORV data, ECA-1.2 would be the correct transition.

Technical Reference(s): E-0 (Rev. 32), step 17
(Attach if not previously provided, ECA-1.2 (Rev. 9), steps 4 and 5
including version/revision number.) ECA-1.2 (Rev. 9), Notes prior to step 1
Proposed references to be provided to applicants during examination: None
Learning
Objective: Discuss conditions which require transition to other procedures from EOP 35 ECA-1.2.
Question Source: Bank #407696
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 82	Tier #		1
K/A Statement:	Group #		2
Knowledge of surveillance procedures related to a	K/A #	APE.028.GEN.2.2.12	
Pzr level control malfunction	Importance Rating		4.1

Proposed Question:

Initial conditions:

- The plant is at 1% power.
- Preparations are being made for entry into MODE 1.
- The time is 23:00.

The following sequence of events occurs:

1. While reviewing the RO rounds sheet, the SM notices the RO did not complete the shiftly surveillance on Pzr level versus Pzr program level.
2. The last shift completed the Pzr level surveillance at 07:00.
3. The operators lose control of Pressurizer level.
4. The crew stabilizes Pzr level at 37%.

Assuming there is no remaining issue with the Pzr Level Control System, answer the following questions **using the attached copy of 3.4.3.1, PRESSURIZER**.

In accordance with section 3/4.0 of Technical Specifications, is the crew allowed to continue the startup into MODE 1? If so, why? If not, what ACTION is the crew required to take?

a) The crew IS allowed to continue the startup, since the Pzr level surveillance time has not exceeded its maximum allowable extension per Surveillance Requirement 4.0.2.
b) The crew IS allowed to continue the startup, since the crew has up to 24 hours to complete the surveillance per Surveillance Requirement 4.0.3.
c) The crew IS NOT allowed to continue the startup per LCO 3.0.4 allowances; but is allowed to remain at 1% power indefinitely while restoring Pzr level and completing the Pzr level surveillance, since LCO 3.4.3.1 is exempt from LCO 3.0.4 shutdown requirements.
d) The crew IS NOT allowed to continue the startup per LCO 3.0.4 requirements; and is required to restore Pzr level within 2 hours, or be in at least HOT STANDBY within the next 6 hours, per LCO 3.4.3.1.

Proposed Answer: D

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to apply the generic LCO and Surveillance requirements of section 3/4.0 of Tech Specs during abnormal plant conditions. The surveillance requirement is to verify Pzr level within +/-6% of program level, which is 28% at 1% power on a shiftly (12 hour) basis. Actual Pzr level is 36%, which does not meet the surveillance requirement of 28% + 6%, = 34%. The ACTION per LCO 3.4.3.1, ACTION b, is to restore level within 2 hours, or be in HOT STANDBY within the next 6 hours.
Surveillance requirement 4.0.1 states that surveillance requirements shall be met during the operational modes or other conditions specified for individual LCOs unless otherwise stated in the individual surveillance requirement. Failure to meet a surveillance, whether such failure is experienced during the performance of the surveillance or between performances of the surveillance, shall be failure to meet the LCO. Failure to perform a surveillance within the specified surveillance interval shall be failure to meet the LCO except as provided in Spec 4.0.3.
Surveillance requirement 4.0.2 states that each Surveillance requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval.
Surveillance requirement 4.0.3 states that if it is discovered that a surveillance was not performed with its specified surveillance interval, then compliance with the requirement to declare the LCO not met may be delayed from the time of discovery, up to 24 hours or up to the limit of the specified surveillance interval, whichever is greater.
LCO 3.0.4 states that entry into an operational mode shall not be made when the conditions of the LCO are not met and the associated action requires a shutdown if they are not met within a time interval.
"A" and "B" are wrong, since a shutdown is required per LCO 3.4.3.1, ACTION b, so, per LCO 3.0.4, entry into an operational mode shall not be made when the conditions of the LCO are not met and the associated action requires a shutdown if they are not met within a time interval.
"A" is plausible, since the Pzr level surveillance was not completed this shift, Pzr level can be quickly restored to within program level requirements per Surveillance 4.4.3.1.1, and Surveillance requirement 4.0.2 allows a 25% time extension to complete a surveillance.
"B" is plausible, since the Pzr level surveillance was not completed this shift, Pzr level can be quickly restored to within program level requirements per Surveillance 4.4.3.1.1, and Surveillance requirement 4.0.3 allows 24 hours to complete a surveillance for surveillances with less than 24 hour frequency.
"C" is wrong, and “D” is correct, since LCO 3.4.3.1 requires a shutdown to HOT STANDBY within the next 6 hours if level is not restored within 2 hours, and LCO 3.0.4 does not allow entry into an operational mode when an LCO is not met and the LCO action requires a shutdown.
"C" is plausible, since LCO 3.0.4 allows entry into an operational mode when conformance with action requirements permits continued operation of the facility for an unlimited period of time, and also allows exceptions to the shutdown requirements when they are specifically stated in the individual specification.

Technical Reference(s):	Tech Spec Section 3/4.0, page 3/4 0-1 (Amend. 213)
(Attach if not previously provided,	Tech Spec Section 3/4.0, pages 3/4 0-2 and 0-3 (Amend. 241)
including version/revision number.)	Tech Spec LCO 3.4.3.1 and Surveillance 4.4.3.1.1 (Amend. 258)
	Tech Spec Figure 3.4-5 (Amend. 242)
	SP3670.1-001 (Rev. 41), page 26

Proposed references to be provided to applicants during examination: **Tech Spec LCO 3.4.3.1**

Learning Objective: Given a plant condition or equipment malfunction (with the Pzr Pressure and Level Control System), use provided reference material to… Evaluate Technical Specification Applicability and determine required Actions.

Question Source:	New
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.43.2, 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 83	Tier #		1
K/A Statement: Ability to determine and interpret	Group #		2
RCS pressure as it applies to a control room evacuation	K/A #	APE.068.AA2.6	
Proposed Question:	Importance Rating		4.3

With the reactor initially at 100% power, a fire is reported in the Cable Spreading Room, resulting in the following sequence of events:

12:00: The crew enters EOP 3509.1 *Control Room, Cable Spreading Area Or Instrument Rack Room Fire*.
12:08: The crew successfully completes the first six Control Room steps in EOP 3509.1, and is looping through the Control Room portion of the procedure.
12:10: The "A" MDAFW Pump spuriously starts.
12:15: The "B" Service Water Pump trips.
12:16: The fire is still burning out of control.
12:30: The crew successfully establishes control at the Aux Shutdown Panel.

In accordance with EOP 3509.1, complete the following statements concerning the time the crew was required to evacuate the Control Room, and the method by which an RCS overpressure condition will be mitigated now that control has been established at the Aux Shutdown Panel.

The crew was first required to evacuate the Control Room at___(1)___. An RCS overpressure condition will be mitigated by the___(2)___per EOP 3509.1.

	(1)	(2)
a)	12:10	Pzr Safety Valves
b)	12:10	"A" PORV
c)	12:15	Pzr Safety Valves
d)	12:15	"A" PORV

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires detailed knowledge of EOP 3509.1 actions that go beyond system knowledge and beyond overall procedure mitigation strategy, and requires the applicant to determine when to proceed ahead to a new procedure section.
"A" and "B" are wrong, since the point at which the crew is required to evacuate the control room is when multiple spurious operations of both trains of equipment occur, and that does not occur until the "B" Service Water Pump starts. "A" and "B" are plausible, since spurious equipment actuation first occurred when the "A" MDAFW pump started.
"C" is wrong, and "D" correct, since at the Aux Shutdown Panel, the crew will be directed to open the "A" PORV block valve and use the "A" PORV for RCS pressure control. "C" is plausible, since all four RCPS have been stopped prior to leaving the Control Room, making Pzr spray unavailable, and the crew closed the PORV block valves prior to leaving the Control Room to protect against an electrical malfunction causing a PORV to spuriously open, resulting in an SIS.

Technical Reference(s): EOP 3509.1 (Rev. 20), steps 1-17, especially steps 2, 4, 10, and 17
(Attach if not previously provided,
including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning
Objective: Given a set of plant conditions, determine the required actions to be taken per EOP 3504.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 84	Tier #		1
K/A Statement:	Group #		2
Knowledge of the bases in Tech Specs for LCOs and	K/A #	APE.076.GEN.2.2.25	
Safety limits associated with high RCS activity	Importance Rating		4.2

Proposed Question:

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

The crew has just logged into LCO 3.4.8, *Specific Activity*, ACTION c, which requires the crew to restore Dose Equivalent I-131 to within limits within 48 hours, or be in HOT STANDBY within 6 hours, and in COLD SHUTDOWN within 36 hours.

Per Technical Specification Bases, complete the following statement.

The crew is required to shut down the plant to minimize the dose consequences of a___(1)___, and the reason they were allowed 48 hours prior to having to shutdown the plant is to ___(2)___.

a) (1) LOCA inside or outside Containment
   (2) allow time for the Degasifier to lower RCS activity levels within Tech Spec limits
b) (1) LOCA inside or outside Containment
   (2) prevent having to shutdown the plant due to an Iodine spike in the RCS
c) (1) Steam Line Break or a SG Tube Rupture
   (2) allow time for the Degasifier to lower RCS activity levels within Tech Spec limits
d) (1) Steam Line Break or a SG Tube Rupture
   (2) prevent having to shutdown the plant due to an Iodine spike in the RCS

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires the applicant to have knowledge of Tech Spec bases related to understanding radiation hazards that may arise during abnormal plant conditions.

"A" and "B" are wrong, since the basis is to minimize the dose consequences of a Steam Line Break or a Steam Generator Tube Rupture. "A" and "B" are plausible, since there are dose release calculations based on Ctmt leakage, and a LOCA outside CTMT would release RCS activity to the environment.

"C" is wrong, since the basis for allowing 48 hours before having to shut down the plant is to allow time for the effects of an Iodine spike to decay away. "C" is plausible, since the Degasifier will remove gaseous activity from the letdown stream before returning the water to the RCS.

"D" is correct, since the basis for the RCS Activity LCO shutdown requirement is to minimize the dose consequences of a Steam Line Break or a Steam Generator Tube Rupture. Also, the completion time of 48 hours is acceptable since it is expected that, if there were an iodine spike, the normal coolant iodine concentration would be restored within this time period. Also, there is a low probability of a SLB or SGTR occurring during this time period

Technical Reference(s): Tech Spec LCO 3.4.8 (Amendment 246), Action c
(Attach if not previously provided, including version/revision number.) T/S Bases for LCO 3.4.8 (LBDR No. MP3-013), page B 3/4 4-5
T/S Bases for LCO 3.4.8 (LBDR No. MP3-013), page B 3/4 4-6a

Proposed references to be provided to applicants during examination: None

Learning Objective: Describe the chemistry requirements associated with the reactor coolant system during the following plant conditions… Mode 1 transient… Mode 1 steady state…

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2, 43.4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 85	Tier #		1
K/A Statement: Ability to determine and interpret	Group #		2
facility conditions and selection of appropriate	K/A #	W/E09 & E10.EA2.1	
procedures during natural circulation operations	Importance Rating		3.8

Proposed Question:

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

T-0:	Offsite power is lost, resulting in a plant trip.
T+10 minutes:	ISO New England reports that power is not expected back for several hours.
T+30 minutes:	A PEO reports a significant leak in the CST.
T+35 minutes:	The STA expresses a concern about having enough combined DWST and CST inventory to reach COLD SHUTDOWN.
T+37 minutes:	The crew enters ES-0.2, *Natural Circulation Cooldown*.
T+45 minutes:	The crew is verifying adequate SHUTDOWN MARGIN exists per ES-0.2, step 2 prior to commencing the cooldown to COLD SHUTDOWN.

Based on current plant conditions, what actions are required to be taken/directed by the US per ES-0.2 to reach COLD SHUTDOWN in the shortest amount of time?

a) Remain in ES-0.2, and commence the cooldown prior to completing the verification of adequate SHUTDOWN MARGIN in step 2 of ES-0.2.
b) Remain in ES-0.2 and cooldown and depressurize the RCS in discrete steps while monitoring Pzr level to ensure the cooldown rate is not excessive.
c) Complete the first 10 steps of ES-0.2, and then transition to ES-0.3, *Natural Circulation Cooldown with Steam Void in Vessel (with RVLMS)*.
d) Transition from ES-0.2, step 2, to ES-0.3, *Natural Circulation Cooldown with Steam Void in Vessel (with RVLMS)*.

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires assessing plant conditions and select entry into an event-specific sub-procedure.
"A" is wrong, since ES-0.2 step 5.a logic does not allow proceeding prior to verifying adequate Shutdown Margin. "A" is plausible, since normally, the crew is allowed to proceed to the next step in an EOP when the current step has been commenced. Also, several EOPs specifically allow proceeding with a cooldown prior to verifying Shutdown Margin, but these EOPs have SIS actuated, which is borating the RCS.
"B" is wrong, since ES-0.2 limits the cooldown rate due to concerns with void formation. "B" is plausible, since it is a misapplication of ES- 0.4 actions, which cools down the plant at the Tech Spec maximum rate, since it is designed to deal with Reactor Vessel Head void formation.
"C" is correct, and "D" wrong, since is the first 10 steps of ES-0.2 are preparatory steps that are required to be completed prior to increasing the cooldown rate in ES-0.3. "D" is plausible, since the appropriate transition is to ES-0.3, and there is a sense of urgency to commence the cooldown.

Technical Reference(s): ES-0.2 (Rev 19-1), steps 2, 5, 9, and Note prior to step 11
(Attach if not previously provided, ES-0.3 (Rev 11-1), Entry Conditions, and step 1
including version/revision number.)

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #408718

Question History: Last NRC Exam Millstone 3 2015 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 # 86	Tier #		2
K/A Statement: Predict impact/mitigate	Group #		1
pressure transient protection during cold shutdown	K/A #	005.A2.2	
on the RHR system	Importance Rating		3.7

Proposed question:

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

- When RCS hot leg temperature decreased to 340°F, the "A" train of RHR was placed in service in the cooldown mode.
- When RCS hot leg temperature decreased to 230°F, the "B" train of RHR was also placed in service in the cooldown mode.
- Hot leg temperatures are currently 220°F.

The following sequence of events occurs:

1. The crew closes the "A" PORV block valve due to a leaky PORV.
2. 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	COPPS is NOT required at this RCS temperature, and one PORV is available
b)	IS	one PORV and one RHR Suction Relief are still available
c)	is NOT	the crew was relying on both RHR suction reliefs to meet COPPS requirements
d)	is NOT	the crew was relying on the two PORVs to meet COPPS requirements

Proposed Answer: B

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.
"A" is wrong, since COPPS is required below 226°F. "A" is plausible, since COPPS is only required at cold temperatures, and the plant isn't in COLD SHUTDOWN.
"B" is correct, 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. COPPS is required <226°F from either 2 PORVs, 2 RHR suction relief valves, or one of each, which is being met.
"C" wrong, since COPPS is armed by procedure when hot leg temperatures decrease to 250°F, and at that time, only one RHR Train was in service. "C" 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. In that case, two RHR suction reliefs provide adequate COPPS protection. Also, the second train of RHR was placed in service prior to being needed per Tech Specs at 220°F.
"D" is wrong, since either 2 PORVs, 2 RHR suction relief valves, or one of each can be credited to meet COPPs requirements. "D" is plausible, since normally, two RHR suction relief valves or two PORVs are available for COPPS, not one of each.

Technical Reference(s): OP 3208 (Rev. 31), Note 3 prior to step 4.3.1, and step 4.3.5
(Attach if not previously provided, OP 3208 (Rev. 31), Notes and Caution prior to step 4.3.34
including version/revision number.) Tech Spec LCO 3.4.9.3 (Amendment 197)
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the major administrative or procedural precautions and limitations placed on the operation of the Residual Heat Removal system, including the basis for each.
Question Source: Bank #317246
Question History: Last NRC Exam Millstone 3 2017 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:	Group #		1
Ability to verify Reactor Protection System alarm	K/A #	012.GEN.2.4.50	
setpoints and operate controls indentified in the ARP	Importance Rating		4.0

Proposed Question:

With the plant at 100% power, the POWER RNG FLUX RATE HI Annunciator illuminates on MB4C, resulting in the following sequence of events:

1. The RO reports the Channel II Flux Rate Hi Reactor Trip Bistable is lit on MB4.
2. Per the ARP, the RO checks the PR NIS indications on MB4, and determines NIS Power Range Channel N42B has failed low.
3. The US directs the RO to remove channel N-42B from input to the AFD and QPTR monitor alarms for computer program 3R5.
4. The RO completes this step 45 minutes after the instrument initially failed.

Complete the following statements concerning the additional action(s) required by the US now that the RO has completed removing the N42B inputs to 3R5; and the action required to reset the POWER RNGE HI FLUX Bistable input to RPS after NI42B is repaired.

The US is required to implement___(1)___, since the associated alarm(s) is/are considered INOPERABLE. After NI42B is repaired, the FLUX RATE HI bistable will be reset by___(2)___.

	(1)	(2)
a)	QPTR surveillances only	selecting RESET at the NIS Drawer
b)	QPTR surveillances only	selecting NORMAL at the 7300 RPS Cabinet
c)	QPTR surveillances and AFD monitoring	selecting RESET at the NIS Drawer
d)	QPTR surveillances and AFD monitoring	selecting NORMAL at the 7300 RPS Cabinet

Proposed Answer: A

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to have knowledge of Tech Spec Surveillance Requirements as they apply to abnormal plant conditions.
"A" is correct, since removing the affected input from the AFD program restores the AFD Monitor Alarm to OPERABLE, so AFD monitoring is not required, while removing the affected input from the QPTR program does not restore OPERABILITY to the QPTR Monitor Alarm, so the US IS required to implement QPTR surveillances. Also, the NIS Hi Rate Bistable is reset at the NIS drawer.
"B" is wrong, since the NIS Hi Rate Bistable is reset at the NIS drawer, not at the RPS Cabinet. "B" is plausible, since the NIS Bistables inputting to RPS are restored by three methods—inserting the Control Power fuses for the affected NIS channel, by selecting NORMAL at the 7300 Cabinet input relay switches, and only the Hi Rate Trip Bistable is reset via a reset switch at the NIS Drawer.
"C" and "D" are wrong, since AFD monitoring is not required, since removing the failed channel restores operability to the AFD Monitor Alarm. "C" and "D" are plausible, since a channel is being removed from inputting to the AFD Alarm, and removing a channel from the QPTR alarm makes it inoperable.

Technical Reference(s): OP 3353.MB4C (Rev. 16), 5-4, steps 2 and 5
(Attach if not previously provided, including version/revision number.) AOP 3571 (Rev. 16), Attachment D, step D.7, including Notes
Tech Spec Surveillance Requirement 4.2.1.1.1 (Amendment 268)
Tech Spec Surveillance Requirement 4.2.4.1 (Amendment 258)

Proposed references to be provided to applicants during examination: None

Learning Objective: For the following conditions, determine the effects on the Nuclear Instrumentation System and on interrelated systems… Power Range instrument failure in mode 1 above P-10

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2 and 43.5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 88	Tier #		2
K/A Statement:	Group #		1
Ability to determine operability and/or availability of	K/A #	022.GEN.2.2.37	
safety related equipment for the CTMT cooling system	Importance Rating		4.6

Proposed Question:

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

1. The Containment Unidentified Leakage Sump Pump (3DAS-P10) is tagged out.
2. The "A" Containment Drains Sump Pump (3DAS-P2A) is off.
3. The "B" Containment Drains Sump Pump (3DAS-P2B) is off.
4. Containment Atmosphere Particulate Monitor (3CMS-RE22) is in service.
5. Containment Atmosphere Gaseous Monitor (3CMS-RE22) is tagged out.
6. The "A" CAR Fan (3HVU-FN1A) is running.
7. The "B" CAR Fan (3HVU-FN1B) is off.
8. The "C" CAR Fan (3HVU-FN1C) is tagged out.
9. The Plant Process Computer and all associated computer points are functioning.

The "A" CAR Fan trips.

Which of the following actions are required to be taken to ensure the minimum required equipment is available to meet RCS Leak Detection Systems OPERABILITY requirements prior to entering MODE 4?

a) Start the "B" CAR Fan; and place AT LEAST one Containment Drains Sump Pump in Automatic.
b) Start the "B" CAR Fan; and place BOTH Containment Drains Sump Pumps in Automatic.
c) Restore the Containment Atmosphere Gaseous Monitor to service; and place AT LEAST one Containment Drains Sump Pump in Automatic.
d) Restore the Containment Atmosphere Gaseous Monitor to service; and place BOTH Containment Drains Sump Pumps in Automatic.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since it requires the applicant to apply Tech Spec Basis knowledge to determine operability.
Restoring operability requires starting a CAR fan to ensure a representative sample is reaching the Radiation Monitor, and, restoring either Unidentified Leakage Sump Pump 3DAS-P10 or either Containment Drains Sump Pump 3DAS-P2A or P2B. Also, the Containment Atmosphere particulate Monitor is required, but not the gaseous monitor.
"A" is correct, since the particulate monitor is already in service, and starting a CAR fan and restoring the "A" Containment Drains Sump Pump meets the leak detection requirement of either the Unidentified Leakage Sump Pump OR either Containment Drains Sump Pump.
"B" is wrong, but plausible, since either Ctmt Unidentified Leakage Sump Pump can be used to meet leak detection requirements. Both are not required.
"C" and "D" are wrong, since the particulate monitor, not the gaseous monitor is required. “C” and “D” are plausible, since one of the two monitors is required (the particulate monitor), and the CAR fan is not specified in Tech Spec bases. Also, a Ctmt Drains Sump Pump is required to be started.

Technical Reference(s): OP 3312B (Rev.8), Prerequisite 2.1.2
(Attach if not previously provided, OP 3335B (Rev.17) Sections 4.1.2 and 4.1.6
including version/revision number.) Tech Spec 3.4.6.1 (Amendment 244)
Tech Spec Bases for LCO 3.4.6.1 (LBDCR No. 07-MP3-032), page B 3/4 4-4 and 4-4a
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the operation of the containment and containment leakage monitoring system under normal, abnormal, and emergency conditions (273041).
Question Source: Bank #402823
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.7, 43.2, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 89	Tier #		2
K/A Statement: Predict impact of conditions which exist for an abnormal shutdown of an RCP in comparison to a normal shutdown of an RCP, and use procedures to mitigate the event	Group #		1
	K/A #	003.A2.2	
	Importance Rating		3.9

Proposed Question:

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

- Reactor Power is 6%.
- The crew is withdrawing Control rods to slowly raise power.

The following sequence of events occur:

1. The RCP HI RANGE LKG FLOW HI annunciator illuminates on MB3B.
2. The RO reports "A" RCP seal return (CBO) flow has increased from 2.3 gpm to 4.3 gpm over the past 3 minutes.
3. The STA reports "A" RCP seal inlet and outlet temperatures have started to slowly increase.
4. The RO reports RCP MID STG INLET PRESS HI Alarm is LIT.
5. The RO reports RCP UP STG INLET PRESS HI Alarm in LIT.

What action is required to be directed by the US?

a) Continue with OP 3203, notify the OMOC, monitor the "A" RCP for further degradation, and request Engineering to evaluate continued pump operation.
b) Transition to OP 3206, *Plant Shutdown*, commence an orderly plant shutdown, and remove the "A" RCP from service when directed in OP 3206.
c) Stop the "A" RCP and go to AOP 3554, *RCP Trip or Stopping a RCP at Power*.
d) Trip the Reactor, stop the "A" RCP, and go to E-0, *Reactor Trip or Safety Injection*.

Proposed Answer: C

Explanation (Optional):
This question is considered SRO level, since applicants are required to assess plant conditions, and select the appropriate procedure and determine the required action based on abnormal conditions.
The conditions provided in the question stem indicate that reactor power is below P-10 (10% power) and that a RCP seal failure is occurring.
"A", "B", and "D" are wrong, and "C" is correct, since per OP3353.MB3B 2-10, step 5, table 1, with CBO flow >4 gpm, with both the RCP MID STG INLET PRESS HI and the RCP UP STG INLET PRESS HI Alarms LIT, step 7 is to be used. With reactor power less than P-10, Step 7.2 directs the crew to stop the affected RCP and go to AOP 3554.
"A" is plausible, since this would be correct if CBO flow was less than 4 gpm per Table 1 and step 9.
"B" is plausible, since this would be correct per Table 1 and step 8 if either the RCP MID STG INLET PRESS HI or the RCP UP STG INLET PRESS annunciators were lit, but not both.
"D" is plausible, since with power above P-10, Table 1 directs the crew to step 7.1, which directs the crew to trip the reactor, stop the RCP, and go to E-0.

Technical Reference(s): OP 3353.MB3B 2-10 (Rev. 14), steps 5 through 9
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: IDENTIFY plant conditions that require entry into AOP 3554.
Question Source: Modified Bank #406855 (Parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 43.5
Comments:
This question is considered "modified", since the power level was changed in the stem, and this change resulted in a new correct answer. Also, old distractor "C" has been changed.

Original Bank Question:

The crew is preparing to synchronize the Main Generator, when the following sequence of events occur:

1. RCP A seal return (CBO) flow increases from 2.3 gpm to 4.3 gpm over 3 minutes.
2. Seal Inlet and outlet temperatures have started to slowly increase.
3. RCP MID STG INLET PRESS HI Alarm is LIT.
4. RCP UP STG INLET PRESS HI Alarm in LIT.

What action is required to be directed by the US?

a) Trip the reactor, Stop RCP A, and go to E-0, Reactor Trip or Safety Injection.
b) Stop RCP A and go to AOP 3554, RCP Trip or Stopping a RCP at Power.
c) Trip the Reactor, Stop RCP A, Close RCP seal return (CBO) isolation valve (CHS-AV8141A), and go to E-0, Reactor Trip or Safety Injection.
d) Refer to OP 3206, Plant Shutdown and commence an orderly plant shutdown.

Correct Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Question # 90	Tier #		2
K/A Statement:	Group #		1
Ability to apply Tech Specs for the	K/A #	062.GEN.2.2.40	
AC electrical distribution system	Importance Rating		4.7

Proposed Question:

The plant is at 100% power, an earthquake occurs, resulting in the following sequence of events:

1. The INVERTER 3 TROUBLE Annunciator is received on MB8.
2. A PEO is dispatched to investigate, and the PEO reports the following at Inverter 3:
   - The "Out of Sync Light" is ON
   - The "Bypass Source Supplying Load" Light is ON
   - The "Inverter Supplying Load" Light is OFF
   - The Battery Charger 3 DC Output Breaker has tripped OPEN.

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

a) 1 hour
b) 2 hours
c) 8 hours
d) 24 hours

Proposed Answer: D

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

Technical Reference(s): Tech Spec LCO 3.8.2.1 (Amendment 258)
(Attach if not previously provided, Tech Spec LCO 3.8.3.1 (Amendment 220)
including version/revision number.) EE-1BA (Rev. 31)
Proposed references to be provided to applicants during examination: **Tech Spec LCOs 3.8.2.1 and 3.8.3.1**
Learning Objective: Describe the major administrative or procedural precautions and limitations associated with the 120 VAC Distribution System, including the basis for each, identified within.. MP3 Technical Specifications…
Question Source: Bank #407009
Question History: Last NRC Exam Millstone 3 2013 NRC Exam
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.8, 43.2
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 91	Tier #		2
K/A Statement: Predict impact of a dropped fuel element on the fuel handling system, and use procedures to mitigate the event	Group #		2
	K/A #	034.A2.1	
	Importance Rating		4.4

Proposed Question:

The plant is in MODE 6 with a refueling in progress, when the following sequence of events occur:

1. The Refueling SRO reports a spent fuel assembly has been dropped in the Spent Fuel Pool.
2. The crew enters EOP 3502, *Fuel Handling Accident*.
3. All personnel are evacuated from the Fuel Building.
4. Per EOP 3502, the crew starts Fuel Building Filter Fan 3HVR*FN10A.

Complete the following statements.

Assuming a normal Spent Fuel Pool level, ___(1)___ percent of iodine gap activity is assumed to be released to the Fuel Building per Tech Spec Basis. EOP 3502 will direct the crew to ___(2)___.

a) (1) one
   (2) start a SLCRS fan
b) (1) one
   (2) initiate Control Building Isolation
c) (1) five
   (2) start a SLCRS fan
d) (1) five
   (2) initiate Control Building Isolation

Proposed Answer: B

Explanation (Optional): This question is considered SRO level, since per 10CFR55.43.7, fuel handling facilities and procedures are SRO topics, as is radiation hazards that may arise during abnormal situations per 10CFR55.43.4. Also, it requires knowledge of Tech Spec Bases per 10CFR55.43.2.
"A" is wrong, since EOP 3502 does not direct the crew to start a SLCRS fan.. "A" is plausible, since the SCLRS fans draw on buildings surrounding Ctmt to limit release of activity from these areas, and radiation alarms in several areas require a SLCRS fan to be started.
"B" is correct, since the Tech Spec Bases for minimum fuel pool level is to remove at least 99% of the assumed iodine gap activity released from a ruptured fuel assembly. Additionally, EOP 3502 directs the crew to initiate CBI for a dropped fuel assembly in the fuel building.
"C" and "D" are wrong, since the Tech Spec Bases for minimum fuel pool level is to remove at least 99% of the assumed iodine gap activity released from a ruptured fuel assembly. "C" and "D" are plausible, since 5% is close to 1%, and multiple TS Basis use 5% in their explanation.

Technical Reference(s): EOP 3502 (Rev. 10), step 1, and steps 10-13
(Attach if not previously provided, including version/revision number.) Tech Spec Bases (LBDR No. 14-MP3-011) for 3/4.9.11

Proposed references to be provided to applicants during examination: None

Learning Objective: Given a set of plant conditions, determine the required actions per EOP 3502.

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Memory or Fundamental Knowledge

10 CFR Part 55 Content: 55.43.2, 43.4, 43.5, 43.7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 92	Tier #		2
K/A Statement:	Group #		2
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation for the area radiation monitoring system	K/A #	072.GEN.2.1.7	
	Importance Rating		4.7

Proposed Question:

With the plant initially at 100% power, indication of a significant fuel failure causes the crew to prepare for a plant shutdown. While the crew prepares, the following sequence of events occurs:

1. A tube rupture occurs on the "A" SG.
2. The RO reports Ctmt radiation has increased to 5 x 10^5 R/hr.
3. The crew enters E-3, *Steam Generator Tube Rupture*.
4. The crew completes cooling down the RCS
5. The RO reports Ctmt radiation has decreased to 5 x 10^4 R/hr, and slowly lowering.
6. The crew opens the Pzr Spray Valves and commences depressurizing the RCS.

The following indications currently exist in the Control Room:

- All RCPs: Running
- "A" SG pressure: 1040 psig and slowly decreasing
- "A" S/G level: 68% NR and slowly decreasing
- Core exit TCs: 400°F and slowly decreasing
- RCS pressure: 1050 psia and slowly decreasing
- PZR level: 9% and slowly increasing
- ECCS injection: 750 gpm and slowly increasing

What action is the crew required to take with the RCS depressurization, and why?

a) STOP the depressurization, since termination criterion for subcooling is met.
b) STOP the depressurization, since RCS Pressure is less than the ruptured SG pressure.
c) Continue the depressurization, since PZR Level is required to be greater than 16%.
d) Continue the depressurization, since PZR Level is required to be greater than 50%.

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since it requires the applicant to determine the effect of a radiation hazard that has arisen in the plant and determine the effect it has on procedure decision points. It also requires the applicant to assess plant conditions, and determine the required action to be taken during emergency conditions which goes beyond system knowledge, and beyond knowledge of the overall strategy of E-3.
Adverse Containment (AC) conditions exist when either Ctmt radiation goes above 10^5 R/hr (it did), or if Ctmt temperature goes above 180°F. If AC is due to high radiation, the crew is required to continue to use AC numbers, even if radiation drops below the AC setpoint. If AC is due to high temperature, the crew discontinues use of AC numbers after temperature drops below the AC setpoint.

"A" is wrong because the termination AC subcooling number is <115°F, and currently, subcooling is 150°F (saturation temperature for 1050 psia 550°F, and the plant is at 400°F). "A" is plausible, since there is a subcooling termination criterion, and this number is significantly elevated during AC conditions.
"B" is wrong, since inadequate Pzr level exists to stop the depressurization. "B" is plausible, since the goal of the depressurization is to get backfill to occur from the ruptured SG into the RCS, and indications of backfill exists. RCS WR pressure (psia) and SG pressure (psig) show RCS pressure is below SG pressure, and backflow is confirmed by SG pressure and level slowly decreasing. The second half of the termination criteria requires Pzr Level >16% (50% AC).
"C" is wrong, since for AC conditions, >50% Pzr level is required to terminate the depressurization. "C" is plausible, since if conditions were not adverse, (and Ctmt radiation as dropped below the AC level), this would be correct.
"D" is correct, since due to adverse Ctmt conditions being met on Ctmt radiation, minimum Pzr level to terminate the depressurization is 50%.

Technical Reference(s): E-0 (Rev. 32), step 5
(Attach if not previously provided, including version/revision number.) E-3 (Rev. 26) Step 16
OP 3272 (Rev. 10), Att. 3, Sheet 2, "Adverse Containment"

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

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per E-3.

Question Source: Modified Bank #407994 (Parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.5, 41.11, 43.4, 43.5

Comments:
This question is considered "modified", since the Containment High Range Area Radiation Monitor readings have been added to the stem, and Pzr level has been raised above the non-adverse Ctmt depressurization termination setpoint. Also, this change makes the old correct answer wrong, so it becomes a new distractor.

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

1. A tube rupture occurs on the "A" SG.
2. The crew enters E-3 *Steam Generator Tube Rupture*.
3. The crew opens the Pzr spray valves and commences depressurization.

The following indications currently exist in the Control Room:

- All RCPs: Running
- "A" SG pressure: 1040 psig Slowly DECREASING
- "A" S/G level: 78% NR Slowly DECREASING
- Core exit TCs: 505°F Slowly DECREASING
- RCS pressure: 1050 psia Slowly DECREASING
- PZR level: 9% Slowly INCREASING
- ECCS injection: 750 gpm Slowly INCREASING

What action is the crew required to take with the RCS depressurization?

a) STOP the depressurization. Termination criterion for subcooling is met.
b) Continue to depressurize until PZR Level is greater than16%, then STOP the depressurization.
c) Continue to depressurize until PZR Level is 73%, then STOP the depressurization.
d) STOP the depressurization. RCS Pressure is less than the ruptured S/G as indicated by backflow.

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Question # 93	Tier #		2
K/A Statement: Ability to interpret control room indications to verify the status and operation of the steam generator system, and understand how operator actions and directives affect plant and system conditions	Group #		2
	K/A #	035.GEN.2.2.44	
	Importance Rating		4.4

Proposed Question:

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

1. The plant trips due to an inadvertent MSI.
2. One Safety Valve on each Steam Generator fails to reseat.
3. The BOP throttles total AFW flow to 530 gpm.

Twenty-five Minutes after the trip, the crew has transitioned to ECA-2.1, *Uncontrolled Depressurization of All Steam Generators* and current conditions are as follows:

- Current procedure: ECA-2.1, step 2, checking AFW flow
- Safety Valves on all four SGs: Indicate closed on MB5
- SG NR Levels: Offscale Low
- SG Pressures: 365 psig and lowering
- RCS Wide range Tcold temperatures: 440°F and lowering
- RCS Thot temperatures: 450°F and lowering

What action, if any, is the crew required to take?

a) Do not adjust Auxiliary Feedwater flow, and continue with ECA-2.1.
b) Transition to E-2, *Faulted Steam Generator Isolation.*
c) Stop AFW flow to three SGs, and feed the fourth SG at maximum rate.
d) Reduce feed flow to a minimum of 100 gpm to each SG.

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since applicants are required to assess plant conditions, determine to remain in the current procedure rather than transition out of it, and select the appropriate action based on abnormal conditions which goes beyond system knowledge and beyond knowing the overall strategy of the Contingency Action procedure.
The RCS has cooled down by about 110°F in 25 minutes. Since cooldown exceeds 80°F/hr, AFW is required to be throttled to a minimum of a 100 gpm per SG ("D" correct).
"A" is wrong, but plausible, since this would be correct if RCS hot leg temperatures were stable.
"B" is wrong, since the safety valve indications are derived from differential temperature switches and may erroneously indicate flow (including indicating closed when flow exists). As such, ECA-2.1 foldout page transitions based on SG pressure increasing, not flow indication. "B" is plausible, since a transition to E-2 will be made if an intact SG becomes available, and also, the crew is at ECA-2.1, step 5, so the foldout page transition criteria apply. It does not apply if SIS is being terminated per steps 11-26.
"C" is wrong, but plausible, since this is a misapplication of the step determining SG feed rate in FR-H.1.
"D" is correct, since the RCS has cooled down by about 110°F in 25 minutes, and with cooldown rate exceeding 80°F/hr, AFW is required to be throttled to a minimum of a 100 gpm per SG.

Technical Reference(s): ECA-2.1 (Rev. 20), step 2 and Foldout page E-2 Transition Criteria
(Attach if not previously provided, E-2 (Rev. 14), step 5, including the Caution
including version/revision number.) FR-H.1 (Rev. 26), step 21.f
Proposed references to be provided to applicants during examination: None
Learning
Objective: Given a set of plant conditions, determine the required actions to be taken per ECA-2.1
Question Source: Modified Bank #407719 (Parent question attached)
Question History: Last NRC Exam N/A
Question Cognitive Level: Comprehension or Analysis
10 CFR Part 55 Content: 55.41.4, 41.5, 43.5
Comments:
This question is considered "modified", since the stem was changed to include information on the low set safety valves showing closed. This change was made to allow replacing distractor B with a new, plausible distractor and provide a K/A match regarding interpretation of control room indications to verify the status of the steam generator system.

Original bank question:

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

1. The plant trips due to an inadvertent MSI.
2. One Safety on each Steam Generator fails to reseat.
3. The BOP throttles total AFW flow to 530 gpm.

Twenty-five Minutes after the trip, current conditions are as follows:

- SG NR Levels: Offscale Low
- RCS Wide range Tcold temperatures: 440°F and decreasing.
- RCS Thot temperatures: 450°F and decreasing.
- The crew has just transitioned to the appropriate Emergency Contingency Action Procedure.

What action will the crew be directed to take with AFW flow?

a) Do not adjust AFW flow, continue with the procedure in effect.
b) Control flow to each Steam Generator as needed to stabilize RCS temperatures.
c) Stop AFW flow to all Steam Generators.
d) Establish 100 gpm AFW flow to each Steam Generator.

Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Question # 94	Tier #		3
K/A Statement:	Group #		1
Knowledge of shift or short-term relief turnover practices	K/A #	GEN.2.1.3	
	Importance Rating		3.9

Proposed Question:

The plant is initially at 100% power, and the following people are in the control room:

- The SM
- The US
- The STA
- The RO
- The BOP
- The Work Control SRO

The following sequence of events occurs:

1. The crew is required to perform an ISO New England requested emergency load reduction per AOP 3575, *Rapid Downpower*.
2. While aligning the EHC insert, the BOP Operator reports feeling light-headed and unable to continue as the BOP operator.

Who is allowed to perform the BOP's manipulations on the Main Boards while the RO continues aligning for boration?

a) The Shift Manager.
b) The Unit Supervisor.
c) The Work Control SRO, as long as direction is given from the Unit Supervisor.
d) The Work Control SRO, as long as permission is given by Senior Operations Management (OMOC or above).

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to have knowledge of administrative procedures that specify implementation requirements during abnormal plant conditions.
OP-AA-100, *Conduct of Operations*, states the following in section 2.3 "Senior Licensed Operators Manipulating Controls": Industry operating experience has shown that senior licensed operator's serving in oversight capacities have unintentionally stepped out of their supervisory roles and manipulated plant controls. These manipulations have contributed to plant events. However, 10CFR55 does authorize a Senior Reactor Operator to manipulate the controls at the facility at which they are licensed. Control Board manipulations will not be performed by the on-watch Unit Supervisor or Shift Manager.
It will not be routine for a senior licensed operator to manipulate the controls, but will be allowed under the following conditions:
1. The senior licensed operator is standing watch as a control room operator (RO / BOP) and the manipulations are required as part of normal watch standing duties (N/A in this case)
or
2. The senior licensed operator is needed to perform the manipulation(s) to assist with a test /surveillance or plant evolution (N/A in this question, since this is an emergent transient with an immediate risk to generation). In these cases, permission must be obtained by Senior Operations Management (OMOC or above) and the manipulation(s) is authorized by the on-watch Unit Supervisor. Additionally, the senior licensed operator is required to sign into the shift narrative log as a control room operator (RO / BOP / Extra RO) or
3. The Unit Supervisor has directed an SRO to perform a manipulation during a transient or an emergent activity that poses an immediate risk to generation ("C" correct, "D" wrong).
"A" and "B" are wrong, since Control Board manipulations will not be performed by the on-watch Unit Supervisor or Shift Manager. "A" and ""B" are plausible as both the US and SM are licensed individuals, there is a necessity to take action, and 10CFR55 does authorize a Senior Reactor Operator to manipulate the controls at the facility at which they are licensed..
"C" is correct, and "D" wrong, since the Unit Supervisor has directed an SRO to perform a manipulation during a transient or an emergent activity that poses an immediate risk to generation. "D" is plausible, since this would be correct if a transient was not in progress.

Technical Reference(s): OP-AP-100 (Rev. 38), Attachment 2, Section 2.3 (page 29)
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: Discuss the short term relief requirements in OP-AA-100 *Conduct of Operations.*
Question Source: Bank #389820
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or fundamental knowledge
10 CFR Part 55 Content: 55.41.10, 43.5
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 95	Tier #		3
K/A Statement:	Group #		2
Ability to perform pre-startup procedures for the facility,	K/A #	GEN.2.2.1	
including operating controls that could affect reactivity	Importance Rating		4.4

Proposed Question:

The crew is preparing to perform a Xenon-free reactor startup per OP 3202, *Reactor Startup*, and initial conditions are as follows:

- Burnup is 15,000 MWD/MTU.
- The selected ECC rod height is Control Bank D at 90 steps.

Prior to commencing the withdrawal of rods, an unobserved dilution event occurs, reducing RCS boron concentration by 100 ppm.

Using the attached Rod Worth and Boron Worth curves, complete the statements below, assuming the ECC was accurate prior to the dilution event occurring.

As criticality is approached, the 1/M plot will predict criticality to occur___(1)___; and in accordance with OP 3202, the crew is required to___(2)___.

a) (1) below RIL
   (2) trip the reactor and enter E-0, *Reactor Trip or Safety Injection*
b) (1) below RIL
   (2) commence immediate boration per AOP 3566, *Immediate Boration*
c) (1) outside of the administrative limit, but above RIL
   (2) insert all control banks back into the core and recalculate the ECC
d) (1) outside of the administrative limit, but above RIL
   (2) continue the startup, and initiate a CR to track the reactivity management event

Proposed Answer: C

Explanation (Optional): This question is considered SRO level, since it requires the applicant to have knowledge of administrative procedures involved in specifying actions based on changes in core reactivity, as well as knowledge of actions based on Tech Spec limits.
"C" is correct, and "A", "B", and "D" wrong, since OP 3209A sets the administrative band is +/- 500 pcm at the estimated time of criticality. OP 3209A sets the target criticality at least 900 pcm above the Rod Insertion Limit. The Rod insertion limit (as identified in the TRM) is "C" Control Bank at 51 steps. In using this information and applying the attached curves (see calculations below), it is identified that criticality is expected or would occur outside of the +/- 500 pcm admin limits but above RIL. When this occurs, OP 3202 step 4.30 directs fully inserting Control Banks and recalculating the ECC (step 4.30).
"A" is plausible since a large dilution has occurred, and OP 3202 directs a Reactor trip for other events (SUR of 1.0 dpm, and uncontrolled cooldown of less than 530°F).
"B" is plausible, since a large dilution has occurred, and OP 3202 directs this action when the reactor is critical (or predicted) below RIL.
"D" is plausible, since as the first part of the answer is correct, and criticality will occur above RIL, satisfying Tech Spec limits.
Calculation: 15,000 MWD/MTU is MOL. IRW at ECP (CBD @ 90 steps) is 580 pcm. IRW at the 500-pcm admin limit is 1080 pcm. IRW at RIL (CBC @ 51 steps) is 1937 pcm. The reactivity difference between RIL and ECC is 1937 pcm - 580 pcm is 1357 pcm. Zero Power DBW is -7.17 pcm/ppm. 1357 pcm / -7.17 pcm/ppm = 189 ppm. The 100 ppm dilution results in a 717 pcm (100 ppm x 7.17 pcm /ppm) reactivity addition (which is outside of the admin limit but above RIL).

Technical Reference(s):	OP 3202 (Rev.24), Steps 3,14, 4.27, and 4.30
(Attach if not previously provided,	OP 3209A (Rev.10), step 4.2.5
including version/revision number.)	U3 TRM (LBDCR 17-MP3-007), Figure 6 on page 8.1-16
	RE Curve and Data Book, Curve RE-D-03 (MP3-18-00, Rev. 0)
	RE Curve and Data Book, Curve RE-F-02 (MP3-18-00, Rev. 0)

Proposed references to be provided to applicants during examination: **Curves RE-D-03 and RE-F-02**

Learning Objective: State the minimum margin between the ECC and RIL, and the basis for that margin

Question Source: Modified Bank #406541 (Parent question attached)

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.1, 41.5, 41.6, 43.5, 43.6

Comments:

This question is considered "modified", since the amount of the dilution was changed to form a new correct answer. Also, distractor "D" was changed to make it more plausible.

Original Bank question:

The crew is preparing to perform a Xenon-free reactor startup per OP 3202 *Reactor Startup*, and initial conditions are as follows:

- Burnup is 15,000 MWD/MTU.
- The selected ECC rod height is Control Bank D at 90 steps.

Prior to commencing the withdrawal of rods, an unobserved dilution event occurs, reducing RCS boron concentration by 250 ppm.

Rod Worth and Boron Worth curves are attached to this exam. As criticality is approached, what will the 1/M plot predict, and in accordance with OP 3202, what action will the crew be required to take to mitigate this event?

a) Criticality will be predicted to occur below RIL. The crew will trip the reactor and enter E-0, *Reactor Trip or Safety Injection*.
b) Criticality will be predicted to occur below RIL. The crew will commence immediate boration per AOP 3566 *Immediate Boration*, and fully insert the control rods into the core.
c) Criticality will be predicted to occur below the administrative limit, but above RIL. The crew will insert all control banks back into the core and recalculate the ECC.
d) Criticality will be predicted to occur below the administrative limit, but above RIL. The crew will continue the startup, but will initiate a CR to track the reactivity management event.

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Question # 96	Tier #		3
K/A Statement:	Group #		2
Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, lineups, tagouts etc.	K/A #	GEN.2.2.15	
	Importance Rating		4.3

Proposed Question:

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

- Charging line flow control valve 3CHS*FCV121 has failed open.
- The crew has determined they can operate the plant by throttling a manual valve closed in-line with 3CHS*FCV121, and aligning 3CHS*HCV190A to provide a "throttleable" charging flow path.
- This configuration is not controlled by an approved procedure.

How are 10CFR50.59, "*Changes, Tests, and Experiments"* safety evaluation requirements required to be addressed?

a) The US will answer questions on OP-AA-100, Attachment 9, "Alternate Plant Configuration Sheet".
b) The US will initiate a CR, and the CR process will address the safety evaluation requirements.
c) The SM will evaluate the effects from the alternate plant configuration on WM-AA-301, *Operational Risk Assessment*.
d) The SM will evaluate the effects and log the results in the Operations narrative log.

Proposed Answer: A

Explanation (Optional): This question is considered SRO level, since the applicant is required to have knowledge of administrative procedures associated with operating changes in the facility.
"A" is correct, since OP-AA-100 requires the US to "Determine if engineering assistance and evaluation under 10 CFR 50.59 is needed by completing Attachment 9" (reference step 12.2 on pg. 80). Attachment 9, Alternate Plant Configuration Sheet, has a list of 7 questions that must be answered. If yes is answered to any question, then engineering assistance and evaluation per 10 CFR 50.59 is required.
"B" is wrong as the Condition Report (CR) process is not specifically used to determine 10CFR 50.59 applicability. "B" is plausible as a CR is often used to help evaluate abnormal or adverse events.
"C" is wrong, since WM-AA-301 does not evaluate or screen for 10 CFR 50.59. "C" is plausible as WM-AA-301 is used to assess and mitigate operational risks associated with plant activities (including emergent activities).
"D" is wrong, since the SM does not evaluate and log results. "D" is plausible as the Shift Managers often log their evaluations in the narrative log.

Technical Reference(s): OP-AA-100 (Rev. 38), pages 80 and 85 (Attachment 9)
(Attach if not previously provided, including version/revision number.) WM-AA-301 (Rev. 20), Table of Contents
PI-AA-200 (Rev. 35) Section 3.3 and Attachment 4
Proposed references to be provided to applicants during examination: None
Learning Objective: Describe the process associated with Alternate Plant Configurations
Question Source: Bank #370369
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10, 43.3
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 97	Tier #		3
K/A Statement:	Group #		3
Ability to control radiation releases	K/A #	GEN.2.3.11	
Proposed Question:	Importance Rating		4.3

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

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

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

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

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

Proposed Answer: A

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

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

Proposed references to be provided to applicants during examination: None

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

Question Source: Bank #404705
Question History: Last NRC Exam Millstone 3 2015 NRC Exam
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.11, 43.4
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:

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

Time	Event
1100 :	The following annunciators are received: • Main Board 2B 3-6A "N-16 High" • Main Board 2B 3-6B "N-16 Alert"
1103:	The US enters AOP 3576, *Steam Generator Tube Leak*.
1104:	The RO opens the N-16 PPC screen and reports 2 out 4 N-16 alarm windows are lit for the "A" Steam Generator

Using the picture of the N-16 screen on the next page of this exam, answer the following questions.

(1) Will AOP 3576 require a plant shutdown? If not, why not?
(2) What action(s) is/are required to be directed by the US per AOP 3576 for this event?

a) (1) No. AOP 3576 will wait for a Chemistry Sample to confirm a leak exists.
   (2) While waiting for the shutdown determination, adjust the radiation monitor setpoints.
b) (1) No. There is a problem with the N-16 monitor. All four N-16 alarm windows for the "A" Steam Generator should be illuminated.
   (2) Call the OMOC and request I&C assistance.
c) (1) Yes.
   (2) Shutdown the plant within 3 hours. Once the reactor is shutdown, go to OP 3208, *Plant Cooldown* to establish COLD SHUTDOWN conditions.
d) (1) Yes.
   (2) Shutdown the plant within 6 hours. Once the reactor is shutdown, remain in AOP 3576 to establish COLD SHUTDOWN conditions.

INSERT PICTURE OF NSS Screen with 'A' SG having a 160 gpd tube leak (only upper and lower ppc screens should be lit)

Proposed Answer: D

Explanation (Optional): This question is considered SRO level, since applicants are required to assess plant conditions, determine whether a shutdown is required, and determine whether or not to remain in the current procedure rather than transition out of it. The applicant is also required to determine the time limit to complete the shutdown, which goes beyond system knowledge and beyond knowing the overall strategy of the procedure.

"A" and "B" are wrong, since with the N-16 High annunciator lit, a unit shutdown is required. "A" is plausible as these are the actions to take below 75 gallons per day leakage. "B" is also wrong since this is the normal expected N16 alarm window pattern less than 30 minutes into an event. The remaining alarms have a 30 and 60 minute calculation period and will not alarm inside of these windows. "B" is plausible since the given leak rate exceeds the dark alarm window setpoints and it is plausible that these alarms should be lit after the time element is met.

"C" is wrong as the reactor shutdown requirement is for 6 hours and not 3 hours. Also, OP 3208 is not used or referenced in AOP 3576. "C" is plausible, since if the Hi Rate alarm was lit, the shutdown time requirement would be 3 hours. OP 3208 has procedure sections to achieve COLD SHUTDOWN and they are used in other EOP's/AOPs. These sections in OP 3208 are identified as "EOP RELATED".

"D" is correct, since AOP 3576 directs a plant shutdown for tube leaks $\geq$ 75 gallons per day. For this leak rate, a six hour shutdown is required and AOP 3576 is a standalone procedure with steps to mitigate the leak and cooldown to COLD SHUTDOWN.

Technical Reference(s):	AOP 3576 (Rev. 8), Section 2.2
(Attach if not previously provided,	AOP 3576 (Rev. 8), steps 1-4, 7, 9, 10, and 18
including version/revision number.)	OP 3208 (Rev. 31), Table of Contents
	OP 3353.MB2B (Rev. 0), 3-6A Setpoint
	OP 3353.MB2B (Rev. 0), 3-6B Setpoint

Proposed references to be provided to applicants during examination: **NSS N-16 Screen**

Learning Objective: Given a set of plant conditions, determine the required actions to be taken per AOP 3576

Question Source: New

Question History: Last NRC Exam N/A

Question Cognitive Level: Comprehension or Analysis

10 CFR Part 55 Content: 55.41.12, 43.4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Question # 99	Tier #		3
K/A Statement:	Group #		4
Knowledge of local auxiliary operator tasks during	K/A #	GEN.2.4.35	
an emergency and the resultant operational effects	Importance Rating		4.0

Proposed Question:

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

1. The crew transitions from E-1, *Loss of Reactor or Secondary Coolant* to ES-1.3, *Transfer to Cold Leg Recirculation*.
2. The crew completes aligning the RCS for cold leg recirculation.

Current conditions are as follows:

- The EOF DSEO reports the EOF and TSC are both fully activated.
- RMT 1 reports plant radiological conditions are acceptable for access to all plant areas outside of Ctmt.
- The RO reports CETCs are slowly decreasing.
- The RO reports ECCS flows are stable at their expected values.
- The US has just directed an operator to startup Spent Fuel Pool Cooling using GA-5.

Complete the following statements concerning which ES-1.3 attachment the US is required to direct the crew to complete, and the reason the attachment is being completed.

ES-1.3,__(1)__ is required to be completed. This attachment will__(2)__.

a) (1) Attachment D, "ESF and Auxiliary Building Sump Pumps"
   (2) direct a PEO to locally turn OFF the ESF Bldg Porous Concrete Groundwater Sump Pump (SRW) to protect its safety related power supply (MCC 32-4T)
b) (1) Attachment D, "ESF and Auxiliary Building Sump Pumps"
   (2) direct a PEO to locally turn OFF the ESF Bldg DAS Sump Pump breakers to allow detection of a leak from the Containment Recirculation System (RSS) piping
c) (1) Attachment E, "In-Vessel Effects Monitoring and Evaluation"
   (2) direct the RO to monitor for indication of Recirculation Sump screen blockage
d) (1) Attachment E, "In-Vessel Effects Monitoring and Evaluation"
   (2) direct the crew to transition to ES-1.4, *Transfer to Hot Leg Recirculation*.

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to assess plant conditions to determine when to implement procedure attachments while performing steps in the body of the procedure.
"A" is wrong, since Attachment D leaves the porous groundwater sump (SRW) pump energized, to protect the Ctmt basemat from the buoyant forces of groundwater. "A" is plausible, since Attachment D is required to be performed at this point in ES-1.3, and it addresses both the DAS sump pumps and the SRW pump. Also, the SRW pump is non-safety related, and it is powered from emergency MCC 32-4T.
"B" is correct, since ES-1.3 directs the crew to perform Attachment D at this point, and this will turn off the DAS Sump Pumps for both the ESF Bldg and the Aux Bldg to allow detection of a passive failure, which is assumed to occur in either the RSS or ECCS systems 24 hours after the LOCA.
"C" and "D" are wrong, since Attachment E is not required to be performed, since the TSC is fully staffed, and the crew entered ES-1.3 from E-1. "C" and "D" are plausible, since Attachment E would be performed if the TSC were not fully activated, and early entry into ECA-1.4 would be considered if the crew had not entered ES-1.3 directly from E-1.

Technical Reference(s):	ES-1.3 (Rev. 18), steps 8, 9, and 12, and Attachments D and E
(Attach if not previously provided,	ES-1.3 Step Deviation Doc (Rev. 18) for steps 9 and 12
including version/revision number.)	ES-1.3 Step Deviation Doc (Rev. 18) for Att. D (pages 6-7)
	ES-1.3 Step Deviation Doc (Rev. 18) for Att. E (pages 7-9)

Proposed references to be provided to applicants during examination: None

Learning Objective:	Given a set of plant conditions, determine the required actions to be taken per ES-1.3 and ES-1.4.
Question Source:	New
Question History:	Last NRC Exam N/A
Question Cognitive Level:	Comprehension or Analysis
10 CFR Part 55 Content:	55.41.8, 43.5
Comments:	

Examination Outline Cross-reference:	Level	RO	SRO
Question # 100	Tier #		3
K/A Statement:	Group #		4
Ability to take action in the emergency plan, including supporting or acting as emergency coordinator if required	K/A #	GEN.2.4.38	
	Importance Rating		4.4

Proposed Question:

The CR-DSEO is responsible for certain activities, which can NOT be delegated, until relieved by the EOF DSEO.

From the list below, select those activities which **CANNOT** be delegated by the CR-DSEO.

1) Authorizing off-site notifications
2) Notify the NRC of a 50.54(x) invocation
3) Conduct the station evacuation
4) Classification of the event
5) Authorizing emergency exposures
6) Performing initial offsite dose assessment

a) 1, 2, 6
b) 1, 4, 5
c) 2, 3, 6
d) 3, 4, 5

Proposed Answer: B

Explanation (Optional):
This question is considered SRO level, since it requires the applicant to have knowledge of administrative procedures regarding implementation of the emergency plan, which is a Shift Manager job function.
Attachment 2 of MP-26-EPI-FAP01 lists the responsibilities for the CR-DSEO which cannot be delegated (until relieved by the EOF DSEO.
"B" is correct, and "A", "C", and "D" wrong, since the list of non-delegable tasks includes: authorizing off-site notifications, classification of the event, and authorizing emergency exposures).
“A”, “C”, and “D” are plausible, since they are important e-plan activities that would be occurring during event mitigation.

Technical Reference(s): MP-26-EPI-FAP01 (Rev. 7), Attachment 2, Sheet 1 of 2
(Attach if not previously provided, including version/revision number.)
Proposed references to be provided to applicants during examination: None
Learning Objective: List the DSEO responsibilities that can’t be delegated by SERO.
Question Source: New
Question History: Last NRC Exam N/A
Question Cognitive Level: Memory or Fundamental Knowledge
10 CFR Part 55 Content: 55.41.10, 43.5
Comments: