Question 1

Plant conditions occurred as follows:

- The unit was at 100%.
- It had been operating for 100 days.
- A loss of off-site power occurred which resulted in a Reactor trip and all control rods inserting.
- The operators are maintaining RCS Tavg at 547°F using the S/G ARV's in automatic control.

1. What would be the approximate decay heat generated (1) second after the trip?

and

2. How will the ARVs be operated over the next (6) hours to hold Tavg at 547°F?

a. 1. 3.5 % RTP
   2. The ARVs auto setpoint must be lowered.

b. 1. 7 % RTP
   2. The ARVs auto setpoint must be raised.

c. 1. 3.5 % RTP
   2. The ARVs auto setpoint must be raised.

d. 1. 7 % RTP
   2. The ARVs auto setpoint must be lowered.

Answer:

b. 1. 7 % RTP
2. The ARV's auto setpoint must be raised.

Explanation:

Incorrect	a.	1. This is the 1 minute expected value per Reactor Theory, Chapter 2, Nuclear Physics, pages 107-109. 2. If S/G pressure is lowered this will also lower Tavg due to Tcold going down due to ARV operations and Thot lowering due to decay heat lowering.
Correct	b.	1. This is the expected value per Reactor Theory, Chapter 2, Nuclear Physics, pages 107-109. 2. As decay heat lowers over time Thot lowers. To keep Tavg at 547°F, S/G pressure must go up.
Incorrect	c.	1. This is the 1 minute expected value per Reactor Theory, Chapter 2, Nuclear Physics, pages 107-109. 2. As decay heat lowers over time Thot lowers. To keep Tavg at 547°F, S/G pressure must go up.
Incorrect	d.	1. This is the expected value per Reactor Theory, Chapter 2, Nuclear Physics, pages 107-109. 2. If S/G pressure is lowered this will also lower Tavg due to Tcold going down due to ARV operations and Thot lowering due to decay heat lowering.

Examination:	RO 2008 Ginna NRC	**Question #:**	1	Rev. 2	Level: RO
Lesson Plan:	RRT02C	Nuclear Physics			
Objective(s):	3.03	Define decay heat.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000007 Reactor Trip - Stabilization – Recovery EK1.04 - Knowledge of the operational implications of the following concepts as they apply to the reactor trip: Decrease in reactor power following reactor trip (prompt drop and subsequent decay)	**ROI:** 3.6	**SROI:** 3.9		
Technical References:	Reactor Theory, Chapter 2, Nuclear Physics, pages 107-109.	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(1)	55.43			

Question 2

Plant conditions are as follows:

- The unit was at 90% power when a Loss of Coolant Accident and a Loss of Off-Site Power occurred.
- Manual Safety Injection (SI) and Containment Isolation (CI) were performed.
- All systems performed as designed.
- RCS pressure is 1510 psig and slowly lowering.
- Pressurizer level is 15 % and rising rapidly.
- Subcooling margin is lowering.

Which of the following is the cause for a these indications?

a. SI accumulator Injection.

b. Pressurizer vapor space leak.

c. SI pumps refilling the Pressurizer.

d. Leak in the Pressurizer variable leg.

Answer:

b. Pressurizer vapor space leak.

Explanation:

Incorrect	a.	RCS pressure is above SI accumulator pressure of ~750#. May choose this due to Pzr level raising.
Correct	b.	These are the classic indications of a vapor space break. Due to the high Rx head temperature and the LOOP (no RCP running and no MFW pump) steam voids form in the Rx head and force water back into the Pzr causing its level to rise.
Incorrect	c.	RCS pressure is above SI pump injection pressure of ~1500#. May choose this due to Pzr level raising.
Incorrect	d.	A leak in the Pressurizer variable leg will make the d/p go up which would indicate a lower Pressurizer level.

Examination:	RO 2008 Ginna NRC	**Question #:**	2	Rev. 2	Level: RO
Lesson Plan:	RTA04C	DECREASE IN REACTOR COOLANT INVENTORY (LOCA)			
Objective(s):	1.01	Discuss the difference in expected indication for a PRZR vapor space break and other Loss of Coolant accidents.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000008 Pressurizer Vapor Space Accident AA2.12 - Ability to determine and interpret the following as they apply to the Pressurizer Vapor Space Accident: PZR level indicators	**ROI:**	3.4	**SROI:**	3.7
Technical References:	LP RTA04C pgs. 6,7.	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 3

Plant conditions are as follows:

- The unit has experienced a large break LOCA and a loss of offsite power.
- Neither Diesel Generator (D/G) started automatically.
- Following a manual start of 1A D/G and manually loading of Bus 14 and 18 per Attachment 8.5, the HCO observed the following:
  - 1A EDG load 2000 kw

Which ONE of the following is the **LONGEST** amount of time the diesel generator can remain at the above conditions and what is correct with regard to D/G loading over time?

a. (.5) hours, D/G loading will lower.

b. (2) hours, D/G loading will remain the same.

c. (.5) hours, D/G loading will remain the same.

d. (2) hours, D/G loading will lower.

Answer:

d. (2) hours, D/G loading will slowly lower.

Explanation:

Incorrect	a.	1. D/G KW Ratings are: 1950 KW continuous, 2250 KW for two (2) hours, 2300 KW for one-half (.5) hours. The time limit is 2 hrs. not .5 hrs. 2. Due to the removal of heat/steam from Cnmt by the CRCFs and CS the load on the D/G will lower by itself. This is a known and expected condition.
Incorrect	b.	1. D/G KW Ratings are: 1950 KW continuous, 2250 KW for two (2) hours, 2300 KW for one-half (.5) hours. 2. FSAR section 8 tables 8.3.2.a and b show D/G loading lowering over time due to lower loading on the CRCFs.
Incorrect	c.	1. D/G KW Ratings are: 1950 KW continuous, 2250 KW for two (2) hours, 2300 KW for one-half (.5) hours. The time limit is 2 hrs. not .5 hrs. 2. FSAR section 8 tables 8.3.2.a and b show D/G loading lowering over time due to lower loading on the CRCFs.
Correct	d.	1. EDG KW Ratings are: 1950 KW continuous, 2250 KW for two (2) hours, 2300 KW for one-half (.5) hours. 2. Due to the removal of heat/steam from Cnmt by the CRCFs and CS the load on the D/G will lower by itself. This is a known and expected condition.

Examination:	RO 2008 Ginna NRC	**Question #:**	3	Rev. 5	Level: RO
Lesson Plan:	R0801C	Diesel Generator System			
Objective(s):	1.01	State the purposes or functions of the Emergency Diesel Generator System.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000011 Large Break LOCA EA1.06 - Ability to operate and monitor the following as they apply to a Large Break LOCA: D/Gs	**ROI:**	4.2	**SROI:**	4.2
Technical References:	LP R0801C, Diesel Generator System pg. 36 FSAR section 8 tables 8.3.2.a and b	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	2004 Ginna NRC Exam		
10CFR 55 Content:	55.41(a)(8)	55.43			

Question 4

Plant conditions are as follows:

- The unit has experienced a Loss of Coolant Accident (LOCA).
- RCS pressure is 1400 psig and trending down rapidly.
- Containment pressure is 8 psig and trending up.
- "A" and "C" SI pumps are running.

1. Which of the conditions given would **FIRST** require the Reactor Coolant Pumps to be secured?

and

2. Why are the Reactor Coolant Pumps secured?

a. 1. When RCS pressure minus maximum S/G pressure is 200 psig.
   2. Mass loss is less if RCPs are turned off, resulting in greater assurance of natural circulation.

b. 1. When RCS pressure minus maximum S/G pressure is 230 psig.
   2. Mass loss is less if RCPs are turned off, resulting in greater assurance of natural circulation.

c. 1. When RCS pressure minus maximum S/G pressure is 200 psig.
   2. Mass loss is greater if RCPs are left running, resulting in possible deeper core uncovery.

d. 1. When RCS pressure minus maximum S/G pressure is 230 psig.
   2. Mass loss is greater if RCPs are left running, resulting in possible deeper core uncovery.

Answer:

d. 1. When RCS pressure minus maximum S/G pressure is 230 psig.
2. Mass loss is greater if RCPs are left running, resulting in possible deeper core uncovery.

Explanation:

Incorrect	a.	Due to adverse conditions RCPs should be secured when the d/p is less than 240 psig so 230 psig is the first correct pressure of the given options. Securing of RCPs at this specific time has no bearing on assuring natural circulation.
Incorrect	b.	Correct given d/p to secure the RCPs. Securing of RCPs at this specific time has no bearing on assuring natural circulation.
Incorrect	c.	Due to adverse conditions RCPs should be secured when the d/p is less than 240 psig so 230 psig is the first correct pressure of the given options.
Correct	d.	Prior to cold leg saturation water is being removed from break and no difference in mass loss for RCP running or not. After cold leg saturation phase separation would allow steam only to be removed with no RCPs where steam and water would be removed with RCPs running RCP trip at this time would lead to deeper core uncovery and higher PCT than if pumps tripped prior to this due to higher mass loss with pump running to this time RCP trip criteria chosen to insure pump trip prior to cold leg saturation. RCS / SG d/p < 210 psig (240 psig adverse containment). Adverse containment conditions are in effect due to Cnmt pressure > 4 psig.

Examination:	RO 2008 Ginna NRC	**Question #:**	4	Rev. 3	Level: RO
Lesson Plan:	RTA04C	DECREASE IN REACTOR COOLANT INVENTORY (LOCA)			
Objective(s):	1.03	Describe the basis for RCP trip criteria to include: 3) Assumption as to loss of RCPs			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000009 Small Break LOCA EK3.23 - Knowledge of the reasons for the following responses as they apply to the small break LOCA: RCP tripping requirements	**ROI:**	4.2	**SROI:**	4.3
Technical References:	E-0, reactor Trip or Safety Injection step 14. Ginna Background Document for E-0 step 14 (RCP TRIP/RESTART in Generic Issues section of Executive Volume).	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 5

Plant conditions are as follows:

- The unit is at 50% power.
- A leak develops that completely drains the reference side of VCT level transmitter LT-112.
- Assume the leak affects only the VCT level transmitter LT-112.

With **NO** operator intervention, how will the plant respond?

Actual VCT will trend . . .

a. up until 83%, because of continuous automatic makeup.

b. down to 20% and then trend up due to automatic makeup.

c. down until the VCT is empty and all running charging pumps lose suction.

d. down to 5% at which time the Charging Pumps will take a suction from the RWST.

Answer:

c. down until the VCT is empty and all running charging pumps lose suction.

Explanation:

Incorrect	a.	If it is assumed that the leak on the reference side of the VCT level transmitter will cause the control system to think there is a low level in the VCT.
Incorrect	b.	LT-112 indication at 20% starts the makeup but since it has failed high there will be no auto makeup.
Correct	c.	With the reference leg at a much lower pressure the indicated level indicates high therefore no makeup will occur and VCT level will drop until Charging Pumps loose suction. LT-112 indication at 20% starts the makeup but since it has failed high there will be no auto makeup.
Incorrect	d.	The logic is 2 out of 2, both LT-112 and LT-139 must both see VCT level at 5% to shift. LT-112 is failed high, logic will not be met.

Examination:	RO 2008 Ginna NRC	**Question #:**	5	Rev. 3	Level: RO
Lesson Plan:	R1601C	Chemical and Volume Control System			
Objective(s):	1.10	Predict the effect(s) of a loss or malfunction of the following component(s) and/or instrumentation on the Chemical and Volume Control System: L) Instrumentation: 6) LT-112			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000022 Loss of Rx Coolant Makeup AA1.08 - Ability to operate and / or monitor the following as they apply to the Loss of Reactor Coolant Makeup: VCT level	**ROI:**	3.4	**SROI:**	3.3
Technical References:	33013-1265-2, Chemical Volume Control Letdown. P-3, Chemical and Volume Control System. Figure 1.	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 6

Plant conditions are as follows:

- The unit is in Mode 5 to support a refueling outage.
- It has been five days since the unit was shutdown from 100% power.
- RCS loops are at 54 inches.
- RCS Temperature is 120°F.
- RHR flow has been stopped due to cavitation to the RHR pumps.

1. What is correct concerning the RCS and the loss of RHR?

and

2. Which one of the following is the preferred method to raise RCS loop level?

a. 1. The RCS is in a reduced inventory condition, core uncovery could occur in a short period of time.
   2. RWST gravity feed.

b. 1. The RCS is in a reduced inventory condition, core uncovery could occur in a short period of time.
   2. Charging to loop B cold leg.

c. 1. The RCS is above the reduced inventory condition, core uncovery times are significantly longer.
   2. RWST gravity feed.

d. 1. The RCS is above the reduced inventory condition, core uncovery times are significantly longer.
   2. Charging to loop B cold leg.

Answer:

a. 1. The RCS is in a reduced inventory condition, core uncovery could occur in a short period of time.
2. RWST gravity feed.

Explanation:

Correct	a.	1. When less than 64" the RCS is in a reduced inventory condition and due to boil off core uncovery will occur. 2. RWST gravity feed is the preferred method to raise RCS loop level per step 10 of AP-RHR.2.
Incorrect	b.	1. When less than 64" the RCS is in a reduced inventory condition and due to boil off core uncovery will occur. 2. Charging to loop B cold leg is the second preferred method to raise RCS loop level per step 11 of AP-RHR.2.
Incorrect	c.	1. When greater than 64" the RCS is not in a reduced inventory condition, times to boiling and core uncovery would be significantly longer. 2. RWST gravity feed is the preferred method to raise RCS loop level per step 10 of AP-RHR.2.
Incorrect	d.	1. When greater than 64" the RCS is not in a reduced inventory condition, times to boiling and core uncovery would be significantly longer. 2. Charging to loop B cold leg is the second preferred method to raise RCS loop level per step 11 of AP-RHR.2.

Examination:	RO 2008 Ginna NRC	**Question #:**	6	Rev. 5	Level: RO
Lesson Plan:	RAP25C	Loss of RHR While Operating at RCS Reduced Inventory Conditions.			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in AP-RHR.2, Loss of RHR While Operating at RCS Reduced Inventory Conditions.			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000025 Loss of RHR System AK1.01 - Knowledge of the operational implications of the following concepts as they apply to Loss of Residual Heat Removal System: Loss of RHRS during all modes of operation	**ROI:**	3.9	**SROI:**	4.3
Technical References:	AP-RHR-2, Loss of RHR While Operating at RCS Reduced Inventory Conditions AP-RHR-1, Loss of RHR	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 7

Plant conditions are as follows:

- The unit has been at 100% power for the last year.
- All systems are in their normal at power alignment.
- Due to a malfunction, Pressurizer Pressure Controller - 431K output begins and continues to slowly rise.

If **NO** operator action is taken, what will the plant response be?

Spray valves initially . . .

a. close, heaters will energize, one PORV will eventually lift.

b. close, heaters will energize, two PORVs will eventually lift.

c. open, Reactor will trip, RCS pressure will eventually stabilize at approximately 1500 psig.

d. open, Reactor will trip, RCS pressure will eventually stabilize at approximately normal operating pressure.

Answer:

c. open, Reactor will trip, RCS pressure will eventually stabilize at approximately 1500 psig.

Explanation:

Incorrect	a.	If the output for controller 431K rises this will result in the Spray valves opening. The heaters would only energize if the output for 431K lowers. The operation of (1) PORV is correct for the given failure as the logic needed to open PORV-431C will not be met due to 431K failing low (3 of 3 required logics to operate the PORV).
Incorrect	b.	If the output for controller 431K rises this will result in the Spray valves opening. The heaters would only energize if the output for 431K lowers. If it is not known or forgotten that the output from 431K goes to PORV-431C, then the operation of (2) PORVs is plausible.
Correct	c.	A sensed low pressure demand would turn off heaters and would open spray valves to drive actual pressure low to meet the setpoint. The spray valves would continue to decrease pressure. A very rapid pressure transient develops which will result in a reactor trip and safety injection with coincident low pressure setpoints. With no operator action, pressure would continue to decrease until SI is initiated, which causes CI and isolates instrument air to containment. Since the spray valves are fail closed valves they would shut, terminating the transient. Pressure will go to approximately the shutoff head for the SI pumps.
Incorrect	d.	Correct operation of the Spray valves however, pressure will not return to normal due to the trip and the continued Pzr spray and subsequent cooldown.

Examination:	RO 2008 Ginna NRC	**Question #:**	7	Rev. 2	Level: RO
Lesson Plan:	R1901C	Pressurizer Pressure and Level Control			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Pressurizer Pressure and Level Control System, or the Reactor Vessel Overpressure Protection System, will have on related plant systems and/or plant operations: d. Plant operation.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000027 Pressurizer Pressure Control System Malfunction AK2.03 - Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: Controllers and positioners	**ROI:**	2.6	**SROI:**	2.8
Technical References:	P-10, Instrument Failure Reference Manual R1901C- Pressurizer Pressure and Level Control	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 8

E-3, Steam Generator Tube Rupture, Step 5 requires that feed to a ruptured steam generator be continued until narrow range level is above a minimum level.

What is the bases for maintaining a minimum level in a ruptured steam generator?

a. To maximize back pressure and minimize break flow.

b. To dilute RCS water with S/G water in anticipation of a release.

c. To prevent dryout and subsequent corrosive failure of additional steam generator U-tubes.

d. To promote thermal stratification in the ruptured SG during the subsequent RCS cooldown and depressurization.

Answer:

d. To promote thermal stratification in the ruptured SG during the subsequent RCS cooldown and depressurization.

Explanation:

Incorrect	a.	Break flow is always a concern during (any LOCA /) SGTR, efforts are made to minimize the leakage into the S/G but is not the bases for the step 5 requirement.
Incorrect	b.	If chosen to recover via Blowdowns this would seem to be a plausible bases but is not the reason for the 7% level requirement per the Background document.
Incorrect	c.	Failure of additional steam generator U-tubes would not be wanted during a SGTR and any action that prevented this from occurring maybe the correct thing to do, but is not the bases for the step in the background document.
Correct	d.	It is also important to maintain the water level in the ruptured steam generator above the top of the U-tubes. When the primary system is cooled in subsequent steps, the steam generator tubes in the ruptured steam generator will approach the temperature of the reactor coolant, particularly if reactor coolant pumps continue to run. If the steam space in the ruptured steam generator expands to contact these colder tubes, condensation will occur which would decrease the ruptured steam generator pressure. This would reduce the reactor coolant subcooling margin and/or increase primary-to-secondary leakage, possibly delaying SI termination or causing SI reinitiation. Consequently, the water level must be maintained above the top of the tubes to insulate the steam space.

Examination:	RO 2008 Ginna NRC	**Question #:**	8	Rev. 3	Level: RO
Lesson Plan:	REP03C	Steam Generator Tube Rupture E-3			
Objective(s):	1.03	State the basis for Cautions, Notes, and Major Action Categories in E-3, Steam Generator Tube Rupture.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	000038 Steam Gen. Tube Rupture 2.4.18 Knowledge of the specific bases for EOPs.	**ROI:**	3.3	**SROI:**	4.0
Technical References:	E-3, SGTR pg. 7 Ginna Background Document for E-3 pg. 38	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	C000.0897		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 9

Plant conditions are as follows:

- A Reactor startup is in progress following a five week Refueling Outage.
- MTC is predicted to be zero at 15% Reactor Power.
- The reactor is critical.
- N 35 - 1x10 -8 amps and stable
- N 36 - 1x10 -8 amps and stable
- Plant startup is on hold to complete required testing.
- A steam line break occurs.
- 5% of rated steam flow is leaking out.

Which one of the following describes the response of the reactor, assuming no reactor trip occurs?

Tavg will trend . . .

a. up. The reactor will go to 5% power.

b. down. The reactor will go subcritical.

c. down. The reactor will go to 5% power.

d. up. The reactor will go subcritical.

Answer:

b. down. The reactor will go subcritical.

Explanation:

Incorrect	a.	With a positive MTC at the beginning of life, as Tavg lowers due to steam demand, negative reactivity will be added driving the Rx subcritcal. Plausible if assumes a (-MTC) and not certain of plant response.
Correct	b.	With a positive MTC at the beginning of life, as Tavg lowers due to steam demand, negative reactivity will be added driving the Rx subcritcal. [Cycle 34, MTC = (+ 0.70)]
Incorrect	c.	With a positive MTC at the beginning of life, as Tavg lowers due to steam demand, negative reactivity will be added driving the Rx subcritcal. Plausible if is forgotten the Rx is at BOL and has a (+ MTC). If it is assumed the Rx is at MOL or EOL with a (-MTC) this answer would be correct.
Incorrect	d.	With a positive MTC at the beginning of life, as Tavg lowers due to steam demand, negative reactivity will be added driving the Rx subcritcal. Plausible if assumes the wrong Tavg response and a (-MTC).

Examination:	RO 2008 Ginna NRC	**Question #:**	9	Rev. 4	Level: RO
Lesson Plan:	RRT05C	Inherent Reactivities			
Objective(s):	2.06	State a typical value for BOL and EOL Moderator Temperature Coefficient ("T, MTC) and explain the variation of Moderator Temperature Coefficient with respect to changes in the following: a. Tavg			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000040 Steam Line Rupture - Excessive Heat Transfer AK1.05 - Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: Reactivity effects of cooldown	**ROI:**	4.1	**SROI:**	4.4
Technical References:	RRT05C, Inherent Reactivities	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(1)	55.43			

Question 10

Plant conditions are as follows:

- The unit is operating at 14% power.
- The turbine is rolling up to 1800 rpm.
- "A" Condensate Pumps is 00S.
- "B" and "C" Condensate Pumps have just tripped.

Which one of the following combinations will occur if no operator action is taken?

	MFPs TRIP:	REACTOR TRIPs on:	TURBINE TRIPs on:
a.	Immediately	Lo-Lo S/G Level	Reactor Trip
b.	Immediately	Turbine Trip	MFW pump Trip
c.	After 60 seconds	Turbine Trip	MFW pump Trip
d.	After 60 seconds	Lo-Lo S/G Level	Reactor Trip

Answer:

d. After 60 seconds Lo-Lo S/G Level Reactor Trip

Explanation:

Incorrect	a.	MFPs trip when <15 psi seal water pressure above Feedwater pump suction pressure (60 second time delay).
Incorrect	b.	MFPs trip when <15 psi seal water pressure above Feedwater pump suction pressure (60 second time delay). MFP trips only cause a Turbine trip when one of the Generator output breakers (1G or 9X) is closed.
Incorrect	c.	Turbine trip less than 50% will not cause the Rx to trip. MFP trips only cause a Turbine trip when one of the Generator output breakers (1G or 9X) is closed.
Correct	d.	MFPs trip when <15 psi seal water pressure above Feedwater pump suction pressure (60 second time delay). With no feed to the S/Gs, the levels will drop and at 17% in 2/3 S/Gs a Rx Trip is generated which in turn generates a Turbine Trip.

Examination:	RO 2008 Ginna NRC	**Question #:**	10	Rev. 3	Level: RO
Lesson Plan:	R4301C	Condensate And Feedwater Systems			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Condensate and Feedwater Systems will have on related plant system(s) and/or plant operations. a. Main Steam System b. Plant operation			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000054 Loss of Main Feedwater AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): Reactor and/or turbine trip, manual and automatic	**ROI:**	4.1	**SROI:**	4.4
Technical References:	AR-D-5, STEAM GEN LO-LO LEVEL LOOP A 2/3 17% AR-H-11, FEED PUMP SEAL WATER LO DIFF PRESS 15 PSI E-0, Rx Trip or SI	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	B059.0010		
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 11

Plant conditions occurred as follows:

1200 – The unit is operating at rated power and all systems in their normal alignment.
1205 – A small Loss of Coolant Accident (LOCA) occurs and the Letdown system is secured per the procedure in an effort to slow the downward trend of Pressurizer water level.
1215 – A Station Blackout occurs (and no Diesel Generators will start).
1225 – Due to a fault, Instrument Bus A is deenergized.
1235 – Battery 1B is dead due to excessive loading.
1245 – An automatic SI actuates due to low Pressurizer pressure.
1250 – The HCO notices the Letdown Isolation Valve-427 is OPEN.

When did the Letdown Isolation Valve-427 OPEN?
(Assume no operator actions have been performed.)

The Letdown Isolation Valve-427 went OPEN shortly after . . .

a. 1215.

b. 1225.

c. 1235.

d. 1245.

Answer:

a. 1215.

Explanation:

Correct	a.	AOV-427 was closed during the isolation of the Letdown system that was given in the stem, due to the loss of the SA/IA compressors on the station blackout AOV-427 went to its failed position-open.
Incorrect	b.	Instrument Bus A does not power AOV-427, but powers one of the channels that feeds into the 2/3 logic to close AOV-427 on low Pzr level. With no power to 2/3 Pzr level channels the logic to close AOV-427 would be satisfied. With the SBO AOV-427 already failed open due the Station blackout. This answer maybe picked if it is believed to be the P/S to AOV-427 and it fails open on a loss of power to the solenoid.
Incorrect	c.	DC MCC 1B does power AOV-427 and its fail position on loss of DC power is open per ER-ELEC.2. AOV-427 already failed open due the Station blackout. This answer maybe picked if it is believed to be the P/S to AOV-427 and it fails open on a loss of power to the solenoid.
Incorrect	d.	The auto SI would generate a CI which would close AOV-427 but with no air compressors running there is no motive force to close AOV-427. The CI would also secure IA to Cnmt which would also remove IA from AOV-427.

Examination:	RO 2008 Ginna NRC	**Question #:**	11	Rev. 3	Level: RO
Lesson Plan:	R1601C	Chemical And Volume Control System			
Objective(s):	1.07	Given system conditions, describe the design features of the Chemical and Volume Control System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Letdown isolation valve AOV-427 on decreasing PZR level of 13% or on containment isolation			
Category:	Tier 1/ Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	000055 Station Blackout EA2.01 - Ability to determine or interpret the following as they apply to a Station Blackout: Existing valve positioning on a loss of instrument air system	**ROI:**	3.4	**SROI:**	3.7
Technical References:	LP R1601C, Chemical And Volume Control System Pg. 15 LP R4701C, Instrument and Service Air System Pg. 22 10905-0717 AOV-427 ER-ELEC.2 pg. 7	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 12

Plant conditions are as follows:

- The unit is at 50% power.
- The Electric plant is in a Normal 50/50 lineup.
- Annunciator L-5, Safeguard Bus Main Breaker Overcurrent Trip just alarmed.

1. What caused the alarm to come in?

and

2. In accordance with Annunciator Response Procedure L-5, which one of the following correctly lists **ALL** the required operator action(s) after the faulted component is identified and isolated from the bus to support restoring power to the affected bus?

a.
1. Bus 18 time rate over 1500 amps.
2. Ensure the associated D/G supply breaker is in Pull stop. Depress the Overcurrent RESET pushbutton for the affected Safeguards bus Normal Feed breaker.

b.
1. Bus 17 time rate over 1500 amps.
2. Ensure associated D/G supply breaker is in Pull stop. Reset the affected Safeguards bus Normal Feed breaker. Depress the Overcurrent RESET pushbutton for the affected Safeguards bus Normal Feed breaker.

c.
1. Bus 14 time rate over 3000 amps.
2. Ensure associated D/G supply breaker is in Pull stop. Reset the affected Safeguards bus Normal Feed breaker. Depress the Overcurrent RESET pushbutton for the affected Safeguards bus Normal Feed breaker.

d.
1. Bus 16 time rate over 3000 amps.
2. Ensure the associated D/G supply breaker is in Pull stop. Depress the Overcurrent RESET pushbutton for the affected Safeguards bus Normal Feed breaker.

Answer:

c. 1. Bus 14 time rate over 3000 amps.
   2. Ensure associated D/G supply breaker is in Pull stop. Reset the affected Safeguards bus Normal Feed breaker. Depress the Overcurrent RESET pushbutton for the affected Safeguards bus Normal Feed breaker.

Explanation:

Incorrect	a.	1. Bus 18 time rate over 1600 amps is the correct rating, not 1500 amps. 2. Must also reset the affected Safeguards bus Normal Feed breaker.
Incorrect	b.	1. Bus 17 time rate over 1600 amps is the correct rating, not 1500 amps. 2. Actions are correct per L-5, Safeguard Bus Main Breaker Overcurrent Trip.
Correct	c.	Amps and actions required per L-5, Safeguard Bus Main Breaker Overcurrent Trip.
Incorrect	d.	1. Bus 16 time rate over 3000 amps is correct. 2. Must also reset the affected Safeguards bus Normal Feed breaker.

Examination:	RO 2008 Ginna NRC	**Question #:**	12	Rev. 4	Level: RO
Lesson Plan:	R0701C	480V Electrical Distribution			
Objective(s):	1.11	Describe indications and/or alarms associated with the 480V Electrical Distribution System that would cause entry into an: a. Annunciator Response Procedure			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	4		
K/A:	000056 Loss of Off-site Power 2.4.50 - Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	**ROI:** 4.2	**SROI:** 4.0		
Technical References:	AR-L-5, Safeguard Bus Main Breaker Overcurrent Trip	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 13

The Control Room Team is restoring power to Instrument Bus "D" while implementing ECA-0.1, Loss of All AC Power Recovery Without SI Required.

1. Where can Instrument Bus "D" be powered from?

and

2. What is the basis for restoring power to Instrument Bus "D"?

a. 1. MCC "A" or MCC "B".
   2. Instrument Bus "D" powers some required CVCS instrumentation.

b. 1. MCC "A" or MCC "D".
   2. Instrument Bus "D" powers some required CVCS instrumentation.

c. 1. MCC "A" or MCC "B".
   2. Instrument Bus "D" powers some required Pressurizer level instrumentation.

d. 1. MCC "A" or MCC "D".
   2. Instrument Bus "D" powers some required Pressurizer level instrumentation.

Answer:

a. 1. MCC "A" or MCC "B".
   2. Instrument Bus "D" powers some required CVCS instrumentation.

Explanation:

Correct	a.	MCC A (Maint. supply) or MCC B (Normal supply). CVCS instrumentation per ECA-0.1 step 4 background document.
Incorrect	b.	MCC "D" powers Instrument Bus "C".
Incorrect	c.	Instrument Bus "D" does not power any Pressurizer level instruments.
Incorrect	d.	MCC "D" powers Instrument Bus "C" and Instrument Bus "D" does not power any Pressurizer level instruments.

Examination:	RO 2008 Ginna NRC	**Question #:**	13	Rev. 2	Level: RO
Lesson Plan:	REC01C	ECA-0.1, Loss of All AC Power Recovery Without SI Required			
Objective(s):	2.01	Given a set of plant and equipment conditions evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement. (ECA-0.1)			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000057 Loss of Vital AC Inst. Bus AK3.01 - Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus	**ROI:**	4.1	**SROI:**	4.4
Technical References:	ER-INST.3, Instrument Bus Power Restoration pg. 11 ECA-0.1, Loss of All AC Power Recovery Without SI Required Ginna Background Document pg. 20	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 14

Plant conditions are as follows:

- The unit has experienced a Loss of All AC power.
- ECA-0.0, Loss of All AC Power has been entered.
- Service Water (SW) Pump switch alignment has been completed per Step 8.

What is the reason for the required Service Water Pump switch alignment called for in this step?

a. To prevent water hammer in the Service Water System when power is restored to the applicable buses.

b. To ensure Containment and ECCS cooling is restored as soon as the AC emergency buses are reenergized.

c. To allow other systems to start and come up to normal pressures and flows prior to starting the SW system and thermally shocking the systems/components being cooled.

d. To allow a Service Water Pump to automatically load on the AC emergency bus to provide Diesel Generator cooling should that associated Diesel Generator start.

Answer:

d. To allow a Service Water Pump to automatically load on the AC emergency bus to provide Diesel Generator cooling should that associated Diesel Generator start..

Explanation:

Incorrect	a.	Water Hammer is a concern but is not referenced in the background document.
Incorrect	b.	Cooling to the Cnmt and ECCS systems is a concern but is not referenced in the background document.
Incorrect	c.	Thermal Shock is a concern but is not referenced in the background document.
Correct	d.	Required per ECA-0.0, step 8 and reason is per the Background document: Defeating the large loads as automatic start, blackout, or SI loading of as many practical is intended to avoid potential overload of the power source. This action permits the operator to evaluate the status of the restored power source and sequence loads onto the bus consistent with bus status and plant conditions. A service water pump is permitted to automatically load on the AC emergency bus to provide D/G cooling should the diesel have started.

Examination:	RO 2008 Ginna NRC	**Question #:**	14	Rev. 3	Level: RO
Lesson Plan:	REC00C	ECA-0.0, Loss of All AC Power			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in ECA-0.0, Loss Of All AC Power.			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000062 Loss of Nuclear Svc Water AK3.03 - Knowledge of the reasons for the following responses as they apply to the Loss of Nuclear Service Water: Guidance actions contained in EOP for Loss of nuclear service water	**ROI:**	4.0	**SROI:**	4.2
Technical References:	ECA-0.0, Loss of All AC Power and applicable Background Document.	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 15

Plant conditions occurred as follows:

- An unisolable Instrument Air header rupture has occurred.
- The unit was tripped and E-0, Reactor Trip or Safety Injection was entered.
- Following the Reactor Trip a Safety Valve on the "A" S/G opened and has failed to completely reclose.
- While executing E-0, the "A" S/G developed a tube rupture.
- E-3, Steam Generator Tube Rupture, is now being performed.
- "A" S/G pressure has fallen to 600 psig.
- No operator controlled cooldown has been started.
- All other systems have function as required.
- All appropriate actions have been performed.

Which one of the following is the correct method used to lower RCS pressure?

a. PRZR PORV using normal supply.

b. PRZR PORV using backup supply.

c. Auxiliary Spray using Charging Pumps.

d. Normal Sprays.

Answer:

b. PRZR PORV using backup supply.

Explanation:

Incorrect	a.	Due to the IA rupture that can't be isolated the normal motive force (air) is not available.
Correct	b.	Nitrogen is available and can be valved in and is correct per RNO step 22a of E-3.
Incorrect	c.	Need IA to open AOV-296 which failed closed on the IA header rupture.
Incorrect	d.	Need IA to open the spray valves which failed closed on the IA header rupture.

Examination:	RO 2008 Ginna NRC	**Question #:**	15	Rev. 2	Level: RO
Lesson Plan:	REP03C	E-3, Steam Generator Tube Rupture			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in E-3, Steam Generator Tube Rupture.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000065 Loss of Instrument Air 2.4.6 - Knowledge of EOP mitigation strategies.	**ROI:**	3.7	**SROI:**	4.7
Technical References:	E-3, Steam Generator Tube Rupture pg. 22 LP R1601C, CVCS pg. 46 LP R1401C, Pzr and PRT pg. 17,18,19	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	B000.0171		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 16

Plant conditions occurred as follows:

- While at rated power, there was a disturbance on the electrical power grid which required the unit to perform a load reduction.
- The unit is now at 85% power.
- RG&E Energy Control Center reported due to the grid disturbance, current is leading voltage and the condition is continuing to degrade.
- Generator loading is 500 MWe

Using the provided Generator Capability Curve, if current conditions continue which Megavar limit will be met **FIRST** and what type of load is the Generator carrying?

a. 1. -159 Megavars 2. Inductive Load

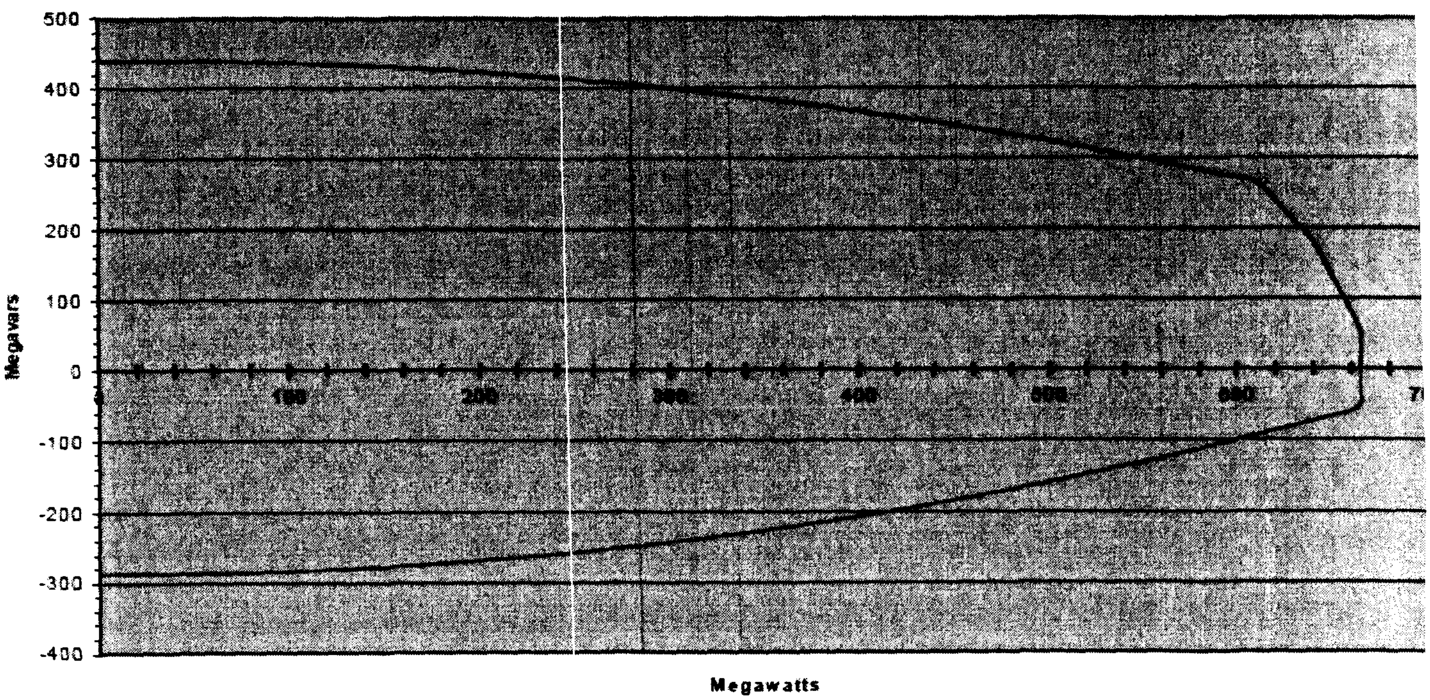
b. 1. 325 Megavars 2. Inductive Load

c. 1. 325 Megavars 2. Capacitive Load

d. 1. -159 Megavars 2. Capacitive Load

GENERATOR CAPABILITY CURVE

19.0 KV 3 phase 60 Hz 1800 RPM
60 psig Hydrogen .92 Power Factor
Rated Cold Gas 46 C



Answer:

d. 1. -159 Megavars 2. Capacitive Load

Explanation:

Incorrect	a.	Wrong type of load.
Incorrect	b.	Wrong side of the curve and wrong type of load.
Incorrect	c.	Wrong side of the curve.
Correct	d.	With current leading voltage it is a Capacitive Load and the Megavar limit can be found on the bottom side of the curve.

Examination:	RO 2008 Ginna NRC	**Question #:**	16	Rev. 3	Level: RO
Lesson Plan:	R5701C	Real and Reactive Load in AC Generators			
Objective(s):	1.02	Recognize the relationships between voltage and current in the following circuits: b. Inductive circuit c. Capacitive circuit			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000077 Generator Voltage and Electric Grid Disturbances AA2.01 - Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: Operating point on the generator capability curve	**ROI:**	3.5	**SROI:**	3.6
Technical References:	LP R5701C, Real and Reactive Load in AC Generators O-6, Operations and Process Monitoring	**References provided during Exam:**	Generator Capability Curve		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 17

Plant conditions are as follows:

- A LOCA outside containment has occurred.
- The crew is performing the actions in ECA-1.2, LOCA Outside Containment.

Which ONE of the following actions will be attempted to isolate the break and which indication is used to determine if the leak has been isolated in accordance with ECA-1.2?

a. Isolate Charging piping then one SI Hot Leg pipe at a time; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

b. Isolate one SI Cold Leg pipe at a time then Charging piping; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

c. Isolate one SI Cold Leg pipe at a time then Charging piping; Pressurizer level is monitored, because with the break isolated, RCS inventory will rapidly rise.

d. Isolate Charging piping then one SI Hot Leg pipe at a time; Pressurizer level is monitored, because with the break isolated, RCS inventory will rapidly rise.

Answer:

b. Isolate one SI Cold Leg pipe at a time then Charging piping; RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.

Explanation:

Incorrect	a.	Charging is isolated in step 4. of ES-1.2, SI Cold Leg (not Hot leg) is isolated in step 3. RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.
Correct	b.	Isolate SI Cold Leg piping per step 3f of ECA-1.2. Charging is isolated in step 4. RCS pressure is monitored, because SI flow will repressurize the RCS with the break isolated.
Incorrect	c.	Isolate SI Cold Leg piping per step 3f of ECA-1.2. Charging is isolated in step 4. Reasonable to conclude this is a viable means of indication that the leak is isolated however not called for in ECA-1.2.
Incorrect	d.	Charging is isolated in step 4. of ES-1.2, SI Cold Leg (not Hot leg) is isolated in step 3. Reasonable to conclude this is a viable means of indication that the leak is isolated however not called for in ECA-1.2.

Examination:	RO 2008 Ginna NRC	**Question #:**	17	Rev. 3	Level: RO
Lesson Plan:	REC12C	ECA-1.2, LOCA Outside Containment			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in ECA-1.2, LOCA Outside Containment.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	W/E04 LOCA Outside Containment EA1.2 - Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment): Operating behavior characteristics of the facility	**ROI:**	3.6	**SROI:**	3.8
Technical References:	ECA-1.2, LOCA Outside Containment pg. 5,6	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	2005 Harris NRC Exam		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 18

Following a unit trip, the plant is in FR-H.1, Response to Loss of Secondary Heat Sink due to a loss of the TDAFW and both MDAFW pumps.

Which of the following lists the proper order for restoring cooling (by priority) as directed by FR-H.1, Response to Loss of Secondary Heat Sink?
(Assume both S/G WR levels are 200 inches and trending down slowly.)

a. SAFW, MFW, condensate, feed and bleed

b. MFW, SAFW, condensate, feed and bleed

c. SAFW, MFW, feed and bleed, condensate

d. MFW, SAFW, feed and bleed, condensate

Answer:

a. SAFW, MFW, condensate, feed and bleed.

Explanation:

Correct	a.	As per FR-H.1 pages 6-16, lists the order as SAFW, MFW, condensate, feed and bleed. By knowing the correct order the correct plant operation is assured.
Incorrect	b.	Recently the correct order was MFW first after AFW then SAFW, this was changed within the last cycle or two.
Incorrect	c.	Condensate, feed and bleed are reversed, this is not allowed by plant procedures and would constitute improper operation of the plant systems if this order of cooling systems were used when both were available.
Incorrect	d.	Recently the correct order was MFW first after AFW then SAFW, this was changed within the last cycle or two. Condensate, feed and bleed are reversed, this is not allowed by plant procedures and would constitute improper operation of the plant systems if this order of cooling systems were used when both were available.

Examination:	RO 2008 Ginna NRC	**Question #:**	18	Rev. 1	Level: RO
Lesson Plan:	RFRH1C	Response To Loss Of Secondary Heat Sink FR-H.1			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in FR-H.1, Response To Loss of Secondary Heat Sink.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink EK2.2 - Knowledge of the interrelations between the (Loss of Secondary Heat Sink) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	**ROI:**	3.9	**SROI:**	4.2
Technical References:	Response To Loss Of Secondary Heat Sink FR-H.1 Pgs. 6-16	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	C000.1261		
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 19

Plant conditions occurred as follows:

- The plant was operating at rated power.
- A control rod dropped.
- The Control Room staff have entered AP-RCC.3, Dropped Rod Recovery.
- Axial Flux Difference (AFD) is no longer within the requirements of the COLR.

In accordance with Technical Specifications, (1) what are the required actions and (2) why?

a. 1. Restore AFD to within limits or be less than 50% Rated Thermal Power in 30 minutes.
   2. To ensure adequate shutdown margin is maintained during recovery of the dropped rod.

b. 1. Restore AFD to within limits or be less than 50% Rated Thermal Power in 60 minutes.
   2. To ensure adequate shutdown margin is maintained during recovery of the dropped rod.

c. 1. Restore AFD to within limits or be less than 50% Rated Thermal Power in 30 minutes.
   2. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses.

d. 1. Restore AFD to within limits or be less than 50% Rated Thermal Power in 60 minutes.
   2. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses.

Answer:

c. 1. Restore AFD to within limits or be less than 50% Rated Thermal Power in 30 minutes.
2. By limiting the amount of power distribution skewing, core peaking factors are consistent with the assumptions used in the safety analyses.

Explanation:

Incorrect	a.	1. Per ITS 3.2.3 action A.1: Restore AFD to within limits or be less than 50% Rated Thermal Power in 30 minutes. 2. Concerns associated with SDM are always valid and should be addressed however, this is not the bases for the LCO.
Incorrect	b.	1. Correct action however the time is limited to 30 minutes. 2. Concerns associated with SDM are always valid and should be addressed however, this is not the bases for the LCO.
Correct	c.	1. Per ITS 3.2.3 action A.1: Restore AFD to within limits or be less than 50% Rated Thermal Power in 30 minutes. 2. Per ITS 3.2.3 action A.1 and associated Bases: The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.
Incorrect	d.	1. Correct action however the time is limited to 30 minutes. 2. Per ITS 3.2.3 action A.1 and associated Bases: The allowed range of the AFD is used in the nuclear design process to confirm that operation within these limits produces core peaking factors and axial power distributions that meet safety analysis requirements.

Examination:	RO 2008 Ginna NRC	**Question #:**	19	Rev. 2	Level: RO
Lesson Plan:	R3001C	ROD CONTROL SYSTEM			
Objective(s):	1.12	Given a set of plant conditions for the Rod Control System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): a. Identify action statements of less than one hour. b. Describe the basis of any (LCO) and/or Safety Limit.			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000003 Dropped Control Rod AK3.05 - Knowledge of the reasons for the following responses as they apply to the Dropped Control Rod: Tech-Spec limits for reduction of load to 50% power if flux cannot be brought back within specified target band	**ROI:**	3.4*	**SROI:**	4.1*
Technical References:	ITS LCO 3.2.3, AFD and associated Bases	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(2)	55.43			

Question 20

Plant conditions occurred as follows:

- With the reactor critical during a startup, Control Rods were stopped at 80 steps withdrawn on Control Bank "D".
- Intermediate Range channel N-35 indicates 1.15E-8 amps and is stable.
- Intermediate Range channel N-36 indicated 1.02E-8 amps and then failed low.
- All Power Range Instruments indicated 2% power.

Which ONE of the following describes the actions to be taken for this situation?

a. Perform NO actions that would result in a positive reactivity addition.

b. Manually trip the Reactor and remain shutdown until N-36 is repaired.

c. Perform a normal Reactor shutdown and remain shutdown until N-36 is repaired.

d. Position the Level Trip switch to BYPASS for N-36 and continue with the startup.

Answer:

d. Position the Level Trip switch to BYPASS for N-36 and continue with the startup.

Explanation:

Incorrect	a.	Not applicable per ITS 3.3.1 R/A E.2, since greater than 5 E-11 amps have 2 hours to be greater than 8% power.
Incorrect	b.	Manual trip not required for these given conditions, , since > 5 E-11 amps have 2 hours to be greater than 8% power or only lower power to < 5 E-11 amps complete shutdown is not required.
Incorrect	c.	Not required per ITS 3.3.1 R/A E.1or 2, since > 5 E-11 amps have 2 hours to be greater than 8% power or only lower power to < 5 E-11 amps complete shutdown is not required.
Correct	d.	Per ITS 3.3.1 R/A E.2 and ER-NIS.2. Place affected IR Level Trip switch to BYPASS and be > 8 % power in (2) hours.

Examination:	RO 2008 Ginna NRC	**Question #:**	20	Rev. 3	Level: RO
Lesson Plan:	R3301C	NUCLEAR INSTRUMENTATION SYSTEM			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Nuclear Instrumentation System will have on related plant system(s) and/or plant operations. a. Rod Control System b. Reactor Protection System			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000033 Loss of Intermediate Range NI 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	**ROI:**	4.3	**SROI:**	4.4
Technical References:	ITS 3.3.1, RTS Instrn. ER-NIS.2, IR Malfunction Pg. 2	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Kewaunee Unit 1 2002 NRC exam.		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 21

Plant conditions occurred as follows:

- The unit was at 100% reactor power.
- Condenser vacuum began to degrade.
- The crew responded using AP-TURB.4, Loss of Condenser Vacuum.
- The crew lowered power as directed.
- The source of Condenser air in-leakage was found and patched but is still leaking.
- Conditions ten minutes later are as follows:
    - Power is now at 75% of full electrical output.
    - Condenser back pressure is 7" Hg absolute and degrading slowly.
    - Annunciator H-7, "Condenser Hi Pressure" is LIT.
    - Annunciator D-24, " Turbine Trip Auto Stop" is NOT LIT.
    - The CRS directs the HCO to trip the reactor.

1. Why was the Reactor Trip directed? and
2. Have the annunciators listed functioned properly?

a. 1. Trip was directed due to operating outside of allowable turbine operating range and was degrading.
   2. Annunciator H-7, "Condenser Hi Pressure" should NOT have alarmed until 22" Hg Condenser vacuum.

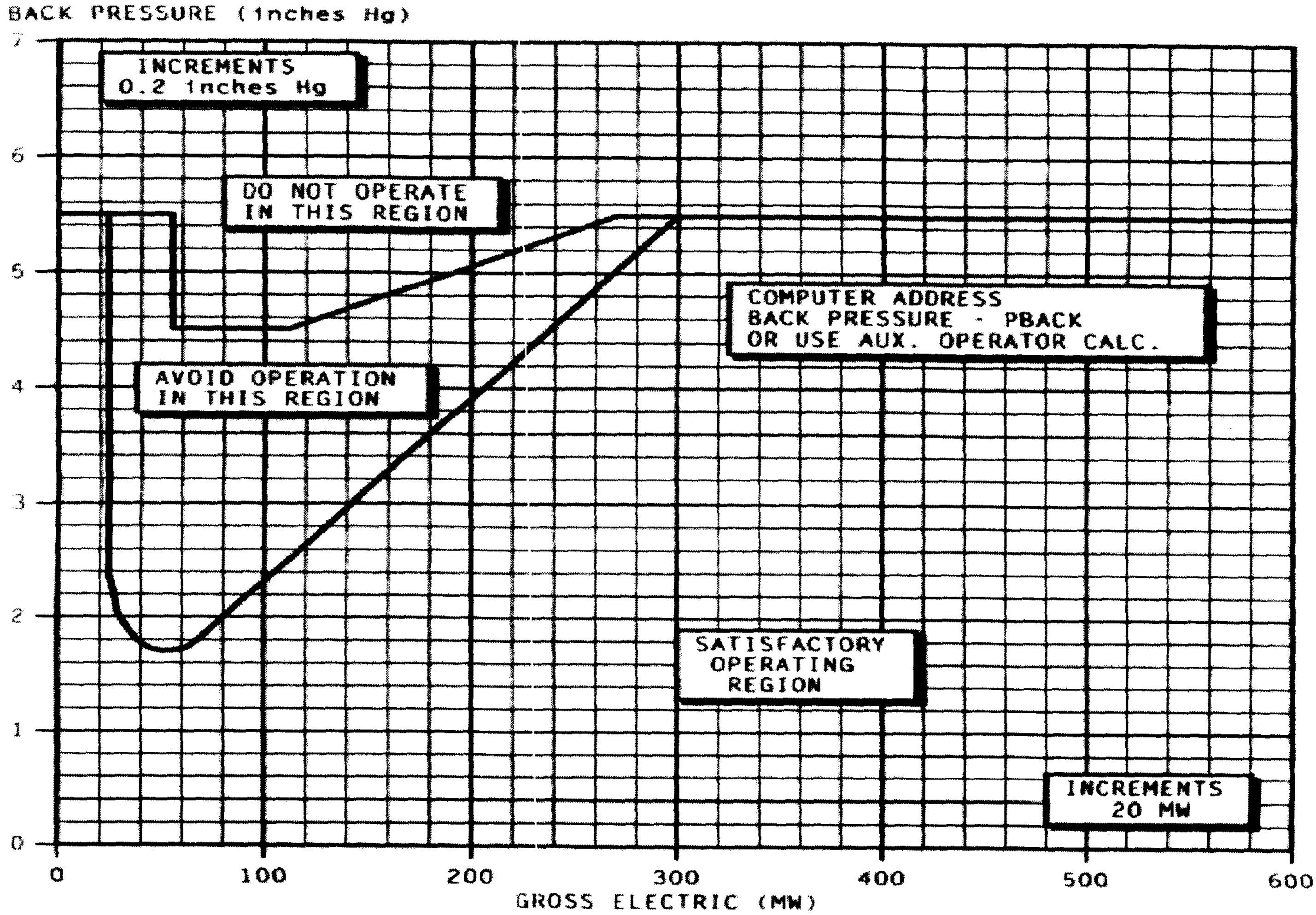
b. 1. Trip was directed since turbine failed to automatically trip.
   2. Annunciator D-24 " Turbine Trip Auto Stop" should have alarmed at 25" Hg Condenser vacuum.

c. 1. Trip was directed due to operating outside of allowable turbine operating range and was degrading towards an automatic trip setpoint.
   2. Annunciators have functioned as designed.

d. 1. Trip was directed since turbine failed to automatically trip.
   2. Annunciators have functioned as designed.

Responsible Manager [signature] Date 5-1-98

FIGURE BACK PRESSURE



Answer:

c. 1. Trip was directed due to operating outside of allowable turbine operating range and was degrading towards an automatic trip setpoint.
2. Annunciators have functioned as designed.

Explanation:

Incorrect	a.	1. Directed per AP-TURB.4 step 2.a RNO. 2. Annunciator H-7, "Condenser Hi Pressure" alarms at 25.5" Hg Condenser vacuum and should be lit.
Incorrect	b.	1. Turbine will not automatically trip until <20" Vacuum. 2. Annunciator D-24 " Turbine Trip Auto Stop" which has many causes to bring in this alarm and one of them is low condenser pressure at 20" Hg, since the condenser is only at ~23" Hg (~7" Hg absolute) the annunciator should not be lit.
Correct	c.	1. Directed per AP-TURB.4 step 2.a RNO. 2. Annunciators alarmed correctly.
Incorrect	d.	1. Turbine will not automatically trip until <20" Vacuum. 2. Annunciators alarmed correctly.

Examination:	RO 2008 Ginna NRC	**Question #:**	21	Rev. 5	Level: RO
Lesson Plan:	RAP23C	Loss of Condenser Vacuum, AP-TURB.4			
Objective(s):	1.03	State the reason/basis for CAUTIONS, NOTES and/or Major Action Categories in AP-TURB.4, Loss of Condenser Vacuum.			
Category:	Tier 1 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000051 Loss of Condenser Vacuum AA2.02 - Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip	**ROI:**	3.9	**SROI:**	4.1
Technical References:	AP-TURB.4 AR D-24 and H-7 Fig-13, Figure Back Pressure LP R4101C, Main Turbine and Turbine Auxs. pg. 42	**References provided during Exam:**	Fig-13, Figure Back Pressure		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

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Question 22

Plant conditions occurred as follows:

- The Control Room has been abandoned due to a fire in the Main Control Board.
- You are the HCO and you are going to the Aux Feed Pump Area to perform applicable sections of ER-FIRE.1, Alternate Shutdown for Control Complex Fire.

Which one of the following correctly lists all the indications at the Intermediate Building Emergency Local Indicating Panel (IBELIP)?

a.	RCS Loop A Thot RCS Loop A Tcold Motor Driven AFW Pump Flow	A and B S/G Pressure A WR S/G Level B WR S/G Level
b.	RCS Loop A Thot RCS Loop A Tcold Turbine Driven AFW Pump Flow	A S/G Pressure A WR S/G Level B WR S/G Level
c.	RCS Loop B Thot RCS Loop B Tcold Turbine Driven AFW Pump Flow	A and B S/G Pressure A WR S/G Level B WR S/G Level
d.	RCS Loop B Thot RCS Loop B Tcold Motor Driven AFW Pump Flow	B S/G Pressure A WR S/G Level B WR S/G Level

Answer:

b.	RCS Loop A Thot RCS Loop A Tcold Turbine Driven AFW Pump Flow	A S/G Pressure A WR S/G Level B WR S/G Level

Explanation:

Incorrect	a.	Motor Driven AFW Pump Flow and B S/G Pressure not available at IBELIP.
Correct	b.	As per ER-FIRE.1, Attachment 3 HCO, step 7.0.
Incorrect	c.	RCS Loop B Thot/Tcold and B S/G Pressure not available at IBELIP.
Incorrect	d.	Motor Driven AFW Pump Flow and B S/G Pressure not available at IBELIP.

Examination:	RO 2008 Ginna NRC	**Question #:**	22	Rev. 1	Level: RO
Lesson Plan:	R4201C	Auxiliary Feedwater System			
Objective(s):	1.02	Given a list of major components or associated instruments of the Auxiliary Feedwater Systems, state their purpose, function, and location.			
Category:	Tier 1 / Group 2	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000068 Control Room Eva. AK2.01 - Knowledge of the interrelations between the Control Room Evacuation and the following: Auxiliary shutdown panel layout	**ROI:**	3.9	**SROI:**	4.0
Technical References:	ER-FIRE.1, Alternate Shutdown for Control Complex Fire.	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 23

Plant conditions are as follows:

- A Steam Line Break has occurred.
- All equipment actuated as required.
- The Control Room staff has isolated the faulted Steam Generator.
- Safety Injection and Containment Isolation have been reset.
- RCS pressure is 1850 psig and slowly trending up.
- There are no other indications of RCS leakage.

Which one of the following describes the sequence and operation of the SI and RHR pumps in accordance with ES-1.1, SI Termination?

a. 1. Establish Charging.
   2. Stop all running SI and RHR pumps.
   3. Verify SI flow not required.

b. 1. Stop all running SI pumps.
   2. Check RCS pressure stable and then establish Charging.
   3. Stop 1 RHR pump. Ensure RCS pressure remains stable, then stop the second RHR pump.

c. 1. Stop 1 SI pump.
   2. Check RCS pressure stable and then establish Charging.
   3. Verify SI flow not required, then stop any running SI pumps and BOTH RHR pumps.

d. 1. Establish Charging.
   2. Check RCS pressure stable and stop 1 SI pump.
   3. Stop 1 RHR pump. Ensure RCS pressure remains stable, then stop any running SI pumps and the second RHR pump.

Answer:

a. 1. Establish Charging.
2. Stop all running SI and RHR pumps.
3. Verify SI flow not required.

Explanation:

Correct	a.	Charging established first, then all pumps stopped, then verification of SI not required per ES-1.1, SI Termination steps 5-8.
Incorrect	b.	Flow reduction sequence not correct. Charging established first, then all pumps stopped, then verification of SI not required per ES-1.1, SI Termination steps 5-8.
Incorrect	c.	Flow reduction sequence not correct. Charging established first, then all pumps stopped, then verification of SI not required per ES-1.1, SI Termination steps 5-8.
Incorrect	d.	Flow reduction sequence not correct. Charging established first, then all pumps stopped, then verification of SI not required per ES-1.1, SI Termination steps 5-8.

Examination:	RO 2008 Ginna NRC	**Question #:**	23	Rev. 2	Level: RO
Lesson Plan:	RES11C	ES-1.1, SI Termination			
Objective(s):	1.04	State the Major Action Categories of ES-1.1, SI Termination.			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	W/E02 SI Termination EA2.2 - Ability to determine and interpret the following as they apply to the (SI Termination): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments	**ROI:**	3.5	**SROI:**	4.0
Technical References:	ES-1.1, SI Termination	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Indian Point Unit 3 2003 NRC Exam		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 24

Plant conditions are as follows:

- The unit has experienced a Loss of Coolant Accident (LOCA).
- All plant equipment operated as designed.
- E-0 and E-1 have been completed.
- While performing ES-1.2, Post LOCA Cooldown and Depressurization, Reactor Coolant System (RCS) conditions are determined to be adequate to support stopping (1) Safety Injection (SI) pump.

When the SI pump is secured, how will the RCS be affected and why?

RCS subcooling will . . .

a. remain the same. Due to reduced SI flow, RCS temperature will rise and with it RCS pressure.

b. remain the same. The flow from the other running SI pumps will rise and achieve a balance with break flow.

c. lower due to lower SI flow. RCS pressure will stabilize at a new lower value when SI flow equals break flow.

d. lower due to break flow remaining constant as SI flow lowers. RCS temperature will rise and stabilizes at a higher value.

Answer:

c. lower due to lower SI flow. RCS pressure will stabilize at a new lower value when SI flow equals break flow.

Explanation:

Incorrect	a.	Subcooling will lower due to lower SI flow and subsequently lower RCS pressure. RCS temperature is controlled by dumping steam.
Incorrect	b.	Subcooling will lower due to lower SI flow and subsequently lower RCS pressure.
Correct	c.	As SI flow lowers due to stopping of (1) SI pump RCS pressure will lower to a new equilibrium value.
Incorrect	d.	RCS temperature is controlled by dumping steam.

Examination:	RO 2008 Ginna NRC	**Question #:**	24	Rev. 1	Level: RO
Lesson Plan:	RES12C	ES-1.2, Post LOCA Cooldown and Depressurization			
Objective(s):	1.02	State the basis for Cautions, Notes, and Major Action Categories of ES-1.2, Post LOCA Cooldown and Depressurization			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	W/E03 LOCA Cooldown - Depress. EA1.2 - Ability to operate and / or monitor the following as they apply to the (LOCA Cooldown and Depressurization): Operating behavior characteristics of the facility	**ROI:**	3.7	**SROI:**	3.9
Technical References:	ES-1.2, Post LOCA Cooldown and Depressurization and Associated Background Document.	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(5)	55.43			

Question 25:

Plant conditions occurred as follows:

- The unit was manually tripped from rated power due to a small-break LOCA.
- Due to an Orange Path on Integrity, FR-P.1, Response to Imminent Pressurized Thermal Shock Condition was entered.
- All SI pumps were stopped.
- Adequate subcooling was verified.
- A soak is required and is in progress.

The Control Room staff is currently performing ES-1.2, Post LOCA Cooldown and Depressurization.

With the soak period still in affect, which one of the following operations can be performed? (Assume any other requirements to perform the step are met.)

a. Lower RCS subcooling by opening a PORV.

b. Fully open an ARV to lower the RCS leak rate.

c. Feed the S/Gs as rapidly as possible to return S/G levels to normal levels.

d. Run SI Pumps as needed to maintain Pressurizer water level above 7%.

Answer:

a. Lower RCS subcooling by opening a PORV.

Explanation:

Correct	a.	This is allowed by FR-P.1 step 29 bases. The "soak" is a period of steady state operation during which any temperature decrease or pressure increase are to be avoided. This time period allows thermal gradients in the reactor vessel wall to be reduced, thus reducing corresponding stresses. Any actions that will not cause either an RCS cooldown or RCS pressure increase and are specified by any other procedure in effect are permitted during this "soak" period.
Incorrect	b.	The leak will have to wait, the important issue is the PRV wall. No cooldown is permitted till the 1 hr soak time is complete.
Incorrect	c.	This would result in a cooldown and is not allowed till the 1 hr soak time is complete.
Incorrect	d.	This would result in a rise in RCS pressure and is not allowed till the 1 hr soak time is complete.

Examination:	RO 2008 Ginna NRC	**Question #:**	25	Rev. 3	Level: RO
Lesson Plan:	RFRP1C	FR-P.1, Response To Imminent Pressurized Thermal Shock Condition			
Objective(s):	1.03	State the basis for Cautions, Notes, and Major Action Categories in FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	W/E08 RCS Overcooling – PTS EK2.2 - Knowledge of the interrelations between the (Pressurized Thermal Shock) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility	**ROI:**	3.6	**SROI:**	4.0
Technical References:	RFRP1C, FR-P.1, Response To Imminent Pressurized Thermal Shock Condition Background Document pgs. 54-56	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 26:

Plant conditions occurred as follows:

- The unit tripped from rated power due to the loss of both RCPs.
- A plant cooldown is underway in accordance with ES-0.2, Natural Circulation Cooldown.
- Current plant conditions are as follows:
  - RCS Pressure – 1550 psig and trending down slowly
  - RCS Tcold – 450°F and trending down
  - F-11, Pressurizer Lo Level 13% has just cleared
  - Reactor Vessel Head Thermocouples - 610°F
  - Pressurizer Level is trending up very quickly from 10%

1. What is the status of the Reactor Coolant System (RCS)?

and

2. What actions are required to be performed?

a. 1 Reactor Head area is subcooled.
   2. Continue cooldown rate in RCS cold legs less than 25°F per hour.

b. 1. Reactor Head area is subcooled.
   2. Raise cooldown rate in RCS cold legs to less than 50°F per hour.

c. 1. Steam voids are forming in the reactor head area.
   2. Repressurize the RCS within allowable limits and continue cooldown.

d. 1. Steam voids are forming in the reactor head area.
   2. Continue with cooldown and stop depressurization.

Answer:

c. 1. Steam voids are forming in the reactor head area.
2. Repressurize the RCS within allowable limits and continue cooldown.

Explanation:

Incorrect	a.	For given set of conditions the Rx head area is not subcooled. This is the correct cooldown for the given conditions.
Incorrect	b.	For given set of conditions the Rx head area is not subcooled. Max cooldown rate is 25°F per hour for given conditions.
Correct	c.	Per ES-0.2, Natural Circulation Cooldown, step 17 – Check for Steam void in Rx head – Unexpected Pzr level variations are seen, then repressurize the RCS within allowable limits and continue cooldown.
Incorrect	d.	Repressurization is required due to steam voiding in the head region.

Examination:	RO 2008 Ginna NRC	**Question #:**	26	Rev. 4	Level: RO
Lesson Plan:	RES02C	ES-0.2, Natural Circulation Cooldown			
Objective(s):	1.02	Given the notes, cautions, and major action categories in ES-0.2, Natural Circulation Cooldown, explain the basis for the same.			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	W/E10 Natural Circ. EK1.3 - Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation with Steam Void in Vessel with/without RVLIS): Annunciators and conditions indicating signals, and remedial actions associated with the Natural Circulation with Steam Void in Vessel with/without RVLIS	**ROI:**	3.3	**SROI:**	3.6
Technical References:	ES-0.2, Natural Circulation Cooldown	**References provided during Exam:**	Steam Tables		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 27

Plant conditions are as follows:

- The reactor has tripped due to a Loss of Coolant Accident (LOCA).
- The Control Room staff is working through the EOPs.
- Due to indications of failed fuel, FR-Z.3, Response To High Containment Radiation Level is entered.

Which of the following are the correct indications and actions per FR-Z.3?

Verify Charcoal filter dampers green status lights are ____1____ and if they are **NOT**, then dispatch personnel to the Relay Room to open the dampers by ____2____ .

a. 1. Extinguished
2. pushing in the trip relay plungers

b. 1. Bright
2. pushing in the trip relay plungers

c. 1. Extinguished
2. opening breakers that supply power to the dampers

d. 1. Bright
2. opening breakers that supply power to the dampers

Answer:

a. 1. Extinguished
2. pushing in the trip relay plungers

Explanation:

Correct	a.	Per FR-Z.3 step 2b and RNO 2b.
Incorrect	b.	Charcoal filter dampers green status lights should be extinguished.
Incorrect	c.	Dampers are opened via plunger actuation not isolation of power. Plausible distracter if candidate equates the green status light being out to no power to the damper, thus failing to its safety configuration.
Incorrect	d.	Charcoal filter dampers green status lights should be extinguished. Dampers are opened via plunger actuation not isolation of power. Plausible distracter if candidate equates the green status light being out to no power to the damper, thus failing to its safety configuration.

Examination:	RO 2008 Ginna NRC	**Question #:**	27	Rev. 1	Level: RO
Lesson Plan:	RFRZ3C	Response to High Containment Radiation Level FR-Z.3			
Objective(s):	2.01	Given a set of plant and equipment conditions evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement. (FR-Z.3)			
Category:	Tier 1 / Group 2	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	W/E16 High Containment Radiation EA1.2 - Ability to operate and / or monitor the following as they apply to the (High Containment Radiation): Operating behavior characteristics of the facility	**ROI:**	2.9	**SROI:**	3.0
Technical References:	FR-Z.3, Response To High Containment Radiation Level pg. 3	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 28

Plant conditions are as follows:

- The unit is at 15% power.
- Annunciator A-15, RCP B CCW RETURN HI TEMP OR LO FLOW 165 GPM 125°F is lit.
- MOV-759B, CCW from RCP 1B ISOL VLV indicates closed and cannot be reopened.
- All RCP seal injection parameters indicate normal.

What is the condition of the plant in (5) minutes, after the Control Room staff completes all of their required actions for these given conditions?

a. "A" RCP is secured, "B" RCP is secured and the unit is at 0% power.

b. "A" RCP is running, "B" RCP is secured and the unit is at 0% power.

c. "A" RCP is secured, "B" RCP is running and the unit is at 0% power.

d. "A" RCP is running, "B" RCP is running and the unit is at 15% power.

Answer:

b. "A" RCP is running, "B" RCP is secured and the unit is at 0% power.

Explanation:

Incorrect	a.	RNO step 4a/b instructs to trip the Rx and only the affected RCP. Want to have (1) running RCP for decay heat removal and better control for plant cooldown.
Correct	b.	Ann A-15 directs the operator to AP-CCW.2 which on RNO step 4a/b instructs to trip the Rx and only the affected RCP.
Incorrect	c.	Correct answer only the RCPs are reversed therefore incorrect.
Incorrect	d.	If it is believed as long as seal injection is supplied to the RCP, the pump can continue to operate which would be incorrect. It is similar to a RCP can loose injection flow as long as CCW cooling is still provided.. AP-CCW.2 RNO step 4a/b instructs to trip the Rx and only the affected RCP.

Examination:	RO 2008 Ginna NRC	**Question #:**	28	Rev.3	Level: RO
Lesson Plan:	R1301C	Reactor Coolant Pump			
Objective(s):	1.10	Predict the effects of a loss or malfunction of the following components and/or instrumentation on the Reactor Coolant Pumps: f. CCW			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	003 Reactor Coolant Pump K6.04 - Knowledge of the effect of a loss or malfunction on the following will have on the RCPS: Containment isolation valves affecting RCP operation	**ROI:** 2.8	**SROI:** 3.1		
Technical References:	A-15, RCP B CCW RETURN HI TEMP OR LO FLOW 165 GPM 125°F AP-CCW.2, Loss of CCW During Power Operations	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 29

Prior to starting a Reactor Coolant Pump (RCP) ensure its Oil Lift Pump has been delivering oil to the ______1______ for at least______2______.

The Oil Lift Pump **WHITE** indicating light will be lit ______3______.

a. 1. upper thrust shoes
   2. two minutes
   3. by a pressure switch when proper lift pressure is attained.

b. 1. lower radial bearing
   2. one minute
   3. as soon as the Oil Lift Pump starts

c. 1. upper thrust shoes
   2. two minutes
   3. as soon as the Oil Lift Pump starts

d. 1. lower radial bearing
   2. one minute
   3. by a pressure switch when proper lift pressure is attained.

Answer:

a. 1. upper thrust shoes
2. two minutes
3. by a pressure switch when proper lift pressure is attained.

Explanation:

Correct	a.	Per precaution 4.5 of S-2.1, Reactor Coolant Pump Operation.
Incorrect	b.	All answers are incorrect per precaution 4.5 of S-2.1, Reactor Coolant Pump Operation.
Incorrect	c.	Oil Lift Pump **WHITE** indicating light will be lit by a pressure switch when proper lift pressure is attained.
Incorrect	d.	Oil Lift Pump must run for at least two minutes.

Examination:	RO 2008 Ginna NRC	**Question #:**	29	Rev. 1	Level: RO
Lesson Plan:	R1301C	REACTOR COOLANT PUMP			
Objective(s):	1.08	Given system conditions, describe the design features of the Reactor Coolant Pumps, to include set points, interlocks, and the related automatic actions for the following components, instrumentation and processes. c. Motor bearing lubrication.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	003 Reactor Coolant Pump A4.03 - Ability to manually operate and/or monitor in the control room: RCP lube oil and lift pump motor controls	**ROI:**	2.8	**SROI:**	2.5
Technical References:	S-2.1, Reactor Coolant Pump Operation	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 30

Plant conditions are as follows:

- The plant has been operating for two weeks following a Refueling outage.
- The reactor is currently at 100% power.
- CCW Surge Tank is 50%.
- VCT level is 25%.
- Rod Control is in automatic.
- Tref and Tavg are matched.
- Tube leaks develop in the Seal Water Return Heat Exchanger.

What will the plant response be if no operator action is taken?

a. VCT level will slowly rise to 100%.

b. CCW Surge Tank will slowly rise to 100%.

c. Rods will step in to maintain Tavg and Tref approximately equal.

d. Rods will step out to maintain Tavg and Tref approximately equal.

Answer:

c. Rods will step in to maintain Tavg and Tref approximately equal.

Explanation:

Incorrect	a.	Level control system for VCT will divert the excess water. Level won't go much higher than ~30%.
Incorrect	b.	RCP #1 Seal Return flow is at a much lower pressure than CCW pressure. CCW Surge Tank level will lower not rise.
Correct	c.	RCP #1 Seal Return flow is at a much lower pressure than CCW pressure. CCW Surge Tank will lower and water will return to the VCT. Since CCW is not borated this is the same as a dilution. As VCT level rises due to the extra water, the VCT divert valve will divert borated return water away from the VCT, resulting in additional dilution. As the RCS is diluted Rx power goes up and Tavg goes up. When Tavg exceeds Tref by 1.5°F rods will step in to lower Tavg.
Incorrect	d.	This response would be correct if there were an inadvertent boration occurring but due to the given conditions a dilution is occurring.

Examination:	RO 2008 Ginna NRC	**Question #:**	30	Rev. 2	Level: RO
Lesson Plan:	R3001C	Rod Control System			
1.07	1.07	Given system conditions, describe the design features of the Rod Control System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Automatic Rod Control			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	004 Chemical and Volume Control K3.01 Knowledge of the effect that a loss or malfunction of the CVCS will have on the following: CRDS (automatic)	**ROI:**	2.5*	**SROI:**	2.9
Technical References:	LP R3001C, Rod Control System, pgs. 25-27 LP R1601C, Chemical and Volume Control System, pgs. 26-29	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(2)	55.43			

Question 31

Plant conditions are as follows:

- The unit is in MODE 5.
- RHR is in shutdown cooling.
- RCS temperature is 180°F and stable.
- An air leak has occurred and as a result Instrument Air has been isolated to the Auxiliary Building.

Provided there is no operator response, how will the plant respond in the next 15 minutes?

a. RCS temperature will rise because HCV-624 and HCV-625 will fail closed and HCV-626 will fail open, therefore providing no cooling.

b. RCS temperature will lower because HCV-624 and HCV-625 will fail open and HCV-626 will fail closed, therefore providing maximum cooling.

c. RCS temperature will remain unchanged because HCV-624, HCV-625 are pinned locally and are unaffected by the loss of Instrument Air.

d. RCS temperature will remain unchanged because CCW supply valves (MOV-738A and MOV-738B) to the RHR Heat Exchangers are manually throttled valves and are unaffected by the loss of Instrument Air.

Answer:

b. RCS temperatue will lower because HCV-624 and HCV-625 will fail open and HCV-626 will fail closed, therefore providing maximum cooling.

Explanation:

Incorrect	a.	This is correct if the failure positions of HCV-624, 625 and 626 were reversed.
Correct	b.	HCV-624 and 625 fail open and HCV-626 will fail closed, therefore providing maximum cooling.
Incorrect	c.	HCV-624, HCV-625 and HCV-626 are not normally operated locally.
Incorrect	d.	MOV-738A and MOV-738B are unaffected by the loss of Instrument Air but RCS temperature will lower because HCV-624 and HCV-625 will fail open and HCV-626 will fail closed, therefore providing maximum cooling.

Examination:	RO 2008 Ginna NRC	**Question #:**	31	Rev. 2	Level: RO
Lesson Plan:	R2501C	Residual Heat Removal System			
Objective(s):	1.10	Predict the effect(s) of a loss or malfunction of the following component(s) and/or instrumentation on the Residual Heat Removal System: a. RHR Heat Exchanger b. Valve 624/625 c. Valve 626			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	005 Residual Heat Removal K6.03 - Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger	**ROI:**	2.5	**SROI:**	2.6
Technical References:	O-2.2, Plant Shutdown from Hot Shutdown to Cold Conditions LP R2501C, Residual Heat Removal System	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 32

Plant conditions are as follows:

- A unit startup is in progress in accordance with O-1.1, Plant Heatup from Cold Shutdown to Hot Shutdown.
- RCS Tcold - 120°F
- RCS pressure - 350 psig
- RCS heatup rate - 25°F per hour
- "A" and "B" RCS Loops are Operable, but only "A" RCP is running.
- The RHR system is aligned for core cooling with "A" and "B" RHR pumps running.
- "B" SI pump is Operable.
- "B" Charging Pump is Operable and providing normal charging flow.

The conditions described above are **IMPROPER** because . . .

a. if the SI pump were to start, it would overpressurize the RCS.

b. the heatup rate is too high for the RCS temperature and pressure.

c. the number of ECCS pumps available to provide injection is inadequate.

d. running one RCP and two RHR pumps produces non-uniform core cooling.

Answer:

a. if the SI pump were to start, it would overpressurize the RCS.

Explanation:

Correct	a.	With the RCS solid this violates ITS 3.4.12, LTOP System, A.1.
Incorrect	b.	Heatup rate is limited to a maximum of 50°F per hour when less than 220°F RCS temperature.
Incorrect	c.	No ECCS pumps are required in Mode 5.
Incorrect	d.	Non-uniform cooling is not the concern for the given conditions.

Examination:	RO 2008 Ginna NRC	**Question #:**	32	Rev. 1	Level: RO
Lesson Plan:	R1001C	Reactor Coolant System			
Objective(s):	1.12	Given a set of plant conditions for the Reactor Coolant System, perform the following in accordance with Technical Specifications, TS Bases, TRM: a. Identify action statements of less than one hour.			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	006 Emergency Core Cooling K5.05 - Knowledge of the operational implications of the following concepts as they apply to ECCS: Effects of pressure on a solid system	**ROI:**	3.4	**SROI:**	3.8
Technical References:	ITS 3.4.12 O-1.1, Plant Heatup from Cold Shutdown to Hot Shutdown, pg. 92	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Cook Unit 1 2002 NRC Exam		
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 33

Plant conditions occurred as follows:

- The plant was operating at 100% power when a Reactor Trip and SI occurred.
- Shortly after the SI actuation the following conditions were observed:
    - R-7, Incore Detector Area radiation monitor high alarm actuated.
    - R-2, Containment radiation monitor is trending up.
    - PRT liquid temperature is 175°F and trending up.
    - Containment pressure and humidity is trending up.
    - "A" Containment Sump level is trending up.
- **NO** operator action has been taken.

Which one of the following would result in these conditions?

a. "B" RCP #1 seal has failed.

b. PCV-435, Pressurizer Safety valve is stuck open.

c. SOV-592, Reactor Head Vent valve has failed open.

d. An Incore Thimble Tube has ruptured at the bottom of the reactor vessel.

Answer:

b. PCV-435, Pressurizer Safety valve is stuck open.

Explanation:

Incorrect	a.	Failure of the "B" RCP seal in conjunction with SI actuation is likely to result in abnormal PRT conditions as the Seal Return Line is isolated (MOV-313), and the excess letdown relief lifts at 150 psig to the PRT. However the amount of flow from a failed # 1 seal is limited and would not result in pressurization of the PRT to rupture disc failure in such a short time.
Correct	b.	Pzr Safety failing open would go to the PRT, rupture the PRT rupture disk at 100 psig then produce the given containment parameters. As the Safety continued to discharge PRT temperature would rise.
Incorrect	c.	SOV-592 is isolated by SV-590 which is normally closed, therefore there would be no release to the containment or the PRT.
Incorrect	d.	Incore thimble tube failure is likely to result in the containment radiation conditions described but will not result in the abnormal PRT conditions.

Examination:	RO 2008 Ginna NRC	**Question #:**	33	Rev. 2	Level: RO
Lesson Plan:	R1401C	Pressurizer and Pressurizer Relief Tank			
Objective(s):	1.02	Given a list of major components or associated instruments of the Pressurizer and Pressurizer Relief Tank System, state their purpose, function, and location (when denoted with L). a. Pressurizer c. Safety Valves (L) e. Pressurizer Relief Tank (L) i. PRT Rupture Disk			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	007 Pressurizer Relief / Quench Tank K3.01 - Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: Containment	**ROI:**	3.3	**SROI:**	3.6
Technical References:	LP R1401C, Pressurizer and Pressurizer Relief Tank pgs. 21,22	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Kewaunee Unit 1 2000 NRC Exam		
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 34

Plant conditions are as follows:

- The unit is at 75% power.
- The electric plant is in an Alternate 50/50 lineup.
- "B" CCW pump is running.
- "B" Diesel Generator is tagged out for maintenance.

1. What is providing power to the running CCW pump?

and

2. If all off-site power is lost, when will the "A" CCW pump start?

a. 1. 7T to Bus 12B to Bus 16.
   2. "A" CCW pump starts 40 seconds after the "A" Diesel Generator ties on to Bus 14.

b. 1. 767 to Bus 12B to Bus 16.
   2. "A" CCW pump starts immediately after the "A" Diesel Generator ties on to Bus 14.

c. 1. 767 to Bus 12B to Bus 16
   2. "A" CCW pump starts 40 seconds after the "A" Diesel Generator ties on to Bus 14.

d. 1. 7T to Bus 12B to Bus 16
   2. "A" CCW pump starts immediately after the "A" Diesel Generator ties on to Bus 14.

Answer:

d. 1. 7T to Bus 12B to Bus 16.
2. "A" CCW pump starts immediately after the "A" Diesel Generator ties on to Bus 14.

Explanation:

Incorrect	a.	Correct power supplies however, CCW pumps starts immediately, Service Water pumps are delayed 40 seconds.
Incorrect	b.	7T is P/S in an Alt 50/50 lineup.
Incorrect	c.	7T is P/S in an Alt 50/50 lineup and CCW pumps starts immediately, Service Water pumps are delayed 40 seconds.
Correct	d.	Correct power supplies. As long as there is no coincident SI signal in conjunction with low/lost voltage, the CCW pumps will immediately start after the "A" Diesel Generator ties on to Bus 14.

Examination:	RO 2008 Ginna NRC	**Question #:**	34	Rev. 2	Level: RO
Lesson Plan:	R2801C	Component Cooling Water System			
Objective(s):	1.05	Identify the electric power supplies to the following system components a. Component Cooling Water Pumps			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	008 Component Cooling Water K2.02 - Knowledge of bus power supplies to the following: CCW pump, including emergency backup	**ROI:**	3.0*	**SROI:**	3.2*
Technical References:	LP, R0601C, 4160VAC Distribution System pgs. 11-17 LP R2801C, Component Cooling Water System Pgs. 18,19 O-6.9.2, pg. 14	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(8)	55.43			

Question 35

Concerning Pressurizer Sprays:

1. What design feature assists in minimizing the Thermal Shock to the Pressurizer Spray Nozzle?

and

2. With spray valves PCV-431A and PCV-431B closed and both RCPs running, what is the **TOTAL** design Pressurizer Spray flow going through the Pressurizer Spray Nozzle?

a. 1. Each Spray line has a hole that is drilled thru the disk of the Spray Valve.
   2. (2) gpm

b. 1. Each Spray line has a bypass line with a manually opened valve.
   2. (2) gpm

c. 1. Each Spray line has a hole that is drilled thru the disk of the Spray Valve.
   2. (4) gpm

d. 1. Each Spray line has a bypass line with a manually opened valve.
   2. (4) gpm

Answer:

b. 1. Each Spray line has a bypass line with a manually opened valve.
2. (2) gpm

Explanation:

Incorrect	a.	Plausible, some plants have this design feature, but not at Ginna. (1) gpm from each spray bypass line = (2) gpm total.
Correct	b.	A manually opened valve in a bypass line provides the flow. (1) gpm from each spray bypass line = (2) gpm total.
Incorrect	c.	Plausible, some plants have this design feature, but not at Ginna. (1) gpm from each spray bypass line = (2) gpm total.
Incorrect	d.	A manually opened valve in a bypass line provides the flow. (1) gpm from each spray bypass line = (2) gpm total.

Examination:	RO 2008 Ginna NRC	**Question #:**	35	Rev. 5	Level: RO
Lesson Plan:	R1401C	Pressurizer and Pressurizer Relief Tank			
Objective(s):	1.04	Describe the (normal cause-effect relationship, functional relationship) between the Pressurizer and Pressurizer Relief Tank System and the given interrelated system a. Reactor Coolant System			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	010 Pressurizer Pressure Control K4.01 - Knowledge of PZR PCS design feature(s) and/or interlock(s) which provide for the following: Spray valve warm-up	**ROI:**	2.7	**SROI:**	2.9
Technical References:	LP R1401C, Pressurizer and Pressurizer Relief Tank Pg. 17	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 36

Plant conditions are as follows:

- The unit was at 100% reactor power.
- Rod control is in Manual and all other controls are in automatic.
- PT-430, Pressurizer Pressure detector is slowly failing high.

1. How will this affect the associated Over Temperature Delta-T trip setpoint?

and

2. Will the associated Over Temperature Delta-T trip setpoint become MORE or LESS Conservative?

a. 1. The trip setpoint lowers.
   2. MORE

b. 1. The trip setpoint lowers.
   2. LESS

c. 1. The trip setpoint rises.
   2. MORE

d. 1. The trip setpoint rises.
   2. LESS

Answer:

d. 1. The Over Temperature Delta-T trip setpoint will rise.
   2. LESS Conservative.

Explanation:

Incorrect	a.	The setpoint does not lower, as pressure goes up so does the set point for OT delta-T. Because the failure causes the setpoint to be higher it is less conservative.
Incorrect	b.	The setpoint does not lower, as pressure goes up so does the set point for OT delta-T. Because the failure causes the setpoint to be higher it is less conservative.
Incorrect	c.	Because the failure causes the setpoint to be higher it is less conservative.
Correct	d.	As pressure goes up so does the set point for OT delta-T. At 100 % power the trip set point would normally be ~80°F with PI-430 failing to 2360 psig the trip set point raises to ~85°F. Therefore the trip associated with this channel would happen at a higher delta-T there by being LESS conservative and putting the core at greater risk.

Examination:	RO 2008 Ginna NRC	**Question #:**	36	Rev. 4	Level: RO
Lesson Plan:	R3501C	Reactor Protection System			
Objective(s):	1.10	Predict the effect(s) of a loss or malfunction of the following component(s) and/or instrumentation on the Reactor Protection System: d. Input (Instrument or Process) Loop Components			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	012 Reactor Protection K6.06 - Knowledge of the effect of a loss or malfunction of the following will have on the RPS: Sensors and detectors	**ROI:**	2.7*	**SROI:**	2.8
Technical References:	LP R3501C, Reactor Protection System pgs. 28,58	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(6)	55.43			

Question 37

Plant conditions occurred as follows:

- The unit was operating at 100% power.
- Normal Letdown was in service.
- A Coolant Leak in the Auxiliary Building occurred.
- The leak caused grounds which lead to an isolation of MCC "D".
- Since the leak could not be isolated a Reactor Trip, a Manual SI and a CI were performed.

At this time, how many Component Cooling Water Containment Penetration Valves (not including any check valves), are closed due to this event?

a. 1

b. 2

c. 3

d. 4

Answer:

b. 2

Explanation:

Incorrect	a.	With the loss of MCC–D only (2) valves will be closed as described in "b" below. If AOV-745 is forgotten, this answer would be selected.
Correct	b.	MOV-813 will get a signal and have power to close. MOV-814 will get a signal but will NOT have power to close. MOV-749A/B and MOV-759A/B do not receive an isolation signal so they will stay open (loss of MCC "D" is not an issue). V-742A is a manual valve and is unaffected. AOV-745 is normally open but gets a CI signal to close.
Incorrect	c.	MOV-813 will get a signal and have power to close. MOV-814 will get a signal but will NOT have power to close, if this is not understood/known it maybe believed this valve will shut. AOV-745 is normally open but gets a CI signal to close.
Incorrect	d.	If it is assumed due to the loss of MCC "D" half of the Cnmt valves won't shut this would be the correct choice however as outlined in "b" above only (2) CCW Containment Penetration Valves will be shut .

Examination:	RO 2008 Ginna NRC	**Question #:**	37	Rev. 3	Level: RO
Lesson Plan:	R2801C	Component Cooling Water System			
Objective(s):	1.04	Describe the (normal cause-effect relationship, functional relationship) between the Component Cooling Water System and the given interrelated system. b. Reactor Coolant Pumps			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	4		
K/A:	013 Engineered Safety Features Actuation K1.08 - Knowledge of the physical connections and/or cause effect relationships between the ESFAS and the following systems: CCWS	**ROI:**	3.6	**SROI:**	3.8
Technical References:	LP R2801C, Component Cooling Water System Pgs. 16,23,24,25 Att.-27.0, Att Auto Action Verification	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 38

Plant conditions are as follows:

- The unit has tripped from rated power due to a large break LOCA.
- Prior to the LOCA, only 1B and 1C Containment Recirc Fans were running and 1D Containment Recirc Fan was in Pull-Stop due to maintenance on its breaker.
- The White Lights for 1B and 1C Containment Recirc Fan Vibrations have just illuminated.
- Annunciator B-32, Containment Recirc Fan Vibration have just alarmed.

What is the status of the Containment Recirc Fans and what operator actions are required?

a. 1A, 1B and 1C Containment Recirc Fans are running.
An AO must be sent to reset the "Vibration Reset" in the Intermediate Building (clean side) and at the Containment Sound Monitor listen for abnormalities.

b. Only 1A Containment Recirc Fan is running.
An AO must be sent to reset the "Vibration Reset" in the Intermediate Building (clean side), then the 1B and 1C Containment Recirc Fans can be manually restarted.

c. 1A, 1B and 1C Containment Recirc Fans are running.
Immediately stop 1B and 1C Containment Recirc Fans, and then send an AO to reset the "Vibration Reset" in the Intermediate Building (clean side), then the 1B and 1C Containment Recirc Fans can be manually restarted and at the Containment Sound Monitor listen for abnormalities.

d. Only 1A Containment Recirc Fan is running.
An AO must be sent to reset the "Vibration Reset" in the Intermediate Building (clean side), then the 1B and 1C Containment Recirc Fan control switches must be placed in stop then fans will automatically restart.

Answer:

a. 1A, 1B and 1C Containment Recirc Fans are running.
An AO must be sent to reset the "Vibration Reset" in the Intermediate Building (clean side) and at the Containment Sound Monitor listen for abnormalities.

Explanation:

Correct	a.	All available fans will start on an SI signal and the Vibration indicators (fan, motor, coupling) will not trip the fans. Actions are required by AR-B-32, Containment Recirc Fan Vibration.
Incorrect	b.	This is correct if the high vibrations were a fan trip but they are not. All available fans 1A, 1B and 1C Containment Recirc Fans will be running. An AO will be sent to reset the "Vibration Reset" in the Intermediate Building (clean side) but this is not required for the fans to run.
Incorrect	c.	All available fans will start on an SI signal and the Vibration indicators (fan, motor, coupling) will not trip the fans. Due to the given conditions even with the high vibrations annunciator in, the fans must be left running and monitored for possible failure. The fan are performing their safety function and must be left on.
Incorrect	d.	This is correct if the high vibrations were a fan trip but they are not. All available fans 1A, 1B and 1C Containment Recirc Fans will be running. An AO will be sent to reset the "Vibration Reset" in the Intermediate Building (clean side) but this is not required for the fans to run.

Examination:	RO 2008 Ginna NRC	**Question #:**	38	Rev. 3	Level: RO
Lesson Plan:	R2201C	Containment, Auxiliary, and Control Building Ventilation Systems			
Objective(s):	1.07	Given system conditions, describe the design features of the Containment, Auxiliary, and Control Building Ventilation Systems, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Air Handling Units and Fans			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	022 Containment Cooling A3.01 - Ability to monitor automatic operation of the CCS, including: Initiation of safeguards mode of operation	**ROI:**	4.1	**SROI:**	4.3
Technical References:	AR-B-32, Containment Recirc Fan Vibration LP R2201C, Containment, Auxiliary, and Control Building Ventilation Systems pgs. 15-18	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 39

Plant conditions are as follows:

- The unit has tripped due to a sudden large break Loss of Coolant Accident (LOCA) coincident with a Loss of all Off-site power.
- All of the Service Water (SW) header isolation valves indicate open.
- FCV-4561 and FCV-4562, CNMT RECIRC fan coolers SW outlet valve status lights are dim.

1. What is the **MINIMUM** number of Service Water header isolation valves that need to be closed to ensure SW can perform its safety function?

and

2. Why are FCV-4561 and FCV-4562 status lights dim?

a. 1. (6) valves
2. The FCVs are full open, they failed full open on the SI signal.

b. 1. (12) valves
2. The FCVs are not fully open, they must be failed full open locally.

c. 1. (12) valves
2. The FCVs are full open, they failed full open on the SI signal.

d. 1. (6) valves
2. The FCVs are not fully open, they must be failed full open locally.

Answer:

d. 1. (6) valves
2. The FCVs are not fully open, they must be failed full open locally.

Explanation:

Incorrect	a.	FCV-4561 and FCV-4562 status lights are bright when failed fully open.
Incorrect	b.	There are (2) valves in each isolation line that is powered from (2) separate trains. As long as any (1) valve in each line closes SW can perform its safety function. FCV-4561 and FCV-4562 status lights are bright when failed fully open.
Incorrect	c.	There are (2) valves in each isolation line that is powered from (2) separate trains. As long as any (1) valve in each line closes SW can perform its safety function.
Correct	d.	There are (2) valves in each isolation line that is powered from (2) separate trains. As long as any (1) valve in each line closes SW can perform its safety function. If the status lights are dim the valves are not full open and the RNO from ATT-27.0 directs the operator to have an AO locally fail the valves full open.

Examination:	RO 2008 Ginna NRC	**Question #:**	39	Rev. 2	Level: RO
Lesson Plan:	R5101C	Service Water System			
Objective(s):	1.07	Given system conditions, describe the design features of the Service Water System, to include set points, interlocks, and the related automatic actions for the following components, instrumentation and/or processes. b. Train separation. e. Engineered Safety Features Actuation System. F. Loop Isolation Motor Operated Valves. g. Temperature Control Valves.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	022 Containment Cooling A1.04 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Cooling water flow	**ROI:**	3.2	**SROI:**	3.3
Technical References:	ATT-27.0 Attachment Automatic Action Verification, pg. 4 LP R5101C, Service Water System Pgs. 29, 30 ITS B3.7.8, pg. B 3.7.8-4	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 40

Plant conditions are as follows:

- The unit has experienced a Design Basis Loss of Coolant Accident (DB LOCA).
- During execution of E-0, a Station Blackout occurred.
- The Control Room Team is performing ECA-0.0, Loss of All AC Power.
- Containment Pressure and Temperature are steadily rising.

1. What is the **MINIMUM** combination of Containment Cooling Systems that must be returned to service to meet the assumptions used in the DB LOCA accident analysis?

and

2. When would the Containment design Temperature limit **FIRST** be exceeded?

a. 1. (1) Containment Spray Pump and (1) Containment Recirc Fan
   2. Containment Temperature > 286°F

b. 1. (1) Containment Spray Pump and (1) Containment Recirc Fan
   2. Containment Temperature > 282°F

c. 1. (1) Containment Spray Pump and (2) Containment Recirc Fans
   2. Containment Temperature > 286°F

d. 1. (1) Containment Spray Pump and (2) Containment Recirc Fans
   2. Containment Temperature > 282°F

Answer:

c. 1. (1) Containment Spray Pump and (2) Containment Recirc Fans
   2. Containment Temperature > 286°F

Explanation:

Incorrect	a.	Need (2) Containment Recirc Fans.
Incorrect	b.	Need (2) Containment Recirc Fans. 282°F is the peak temperature expected following a DB LOCA.
Correct	c.	IAW ITS 3.6.6, CS/CRFC Bases: During a DBA, a minimum of 2 CRFC units and one CS train are required to maintain the containment peak pressure and temperature below the design limits (Ref. 8). To ensure that these requirements are met, two CS trains, four CRFC units, and the NaOH System must be OPERABLE. Therefore, in the event of an accident, at least one CS train, the NaOH System, and two CRFC units operate, assuming the worst case single active failure occurs. Containment Temperature Limit is 286°F.
Incorrect	d.	282°F is the peak temperature expected following a DB LOCA.

Examination:	RO 2008 Ginna NRC	**Question #:**	40	Rev. 1	Level: RO
Lesson Plan:	R2101C	CONTAINMENT AND CONTAINMENT ISOLATION SYS			
Objective(s):	1.07	Given system conditions, describe the design features of the Containment and Containment Isolation System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Containment.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	026 Containment Spray A1.02 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCS controls including: Containment temperature	**ROI:** 3.6	**SROI:** 3.8		
Technical References:	LP R2101C, pgs. 8,9 ITS 3.6.6, CS/CRFC Bases pg. B 3.6.6-6	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 41

Plant conditions are as follows:

- The unit is performing a startup following a Mid-cycle maintenance outage.
- The SHUTDOWN BANKS have been withdrawn.
- The CONTROL BANKS are being withdrawn.
- ARVs are in Manual.

What plant evolution would result in the reactor going critical **PRIOR** to the expected value obtained in accordance with the Estimated Critical Position (ECP)? (Assume no other operator action.)

a. Steam Generator operating level is reduced from 60% to 50%.

b. (1) Steam Generator Automatic Relief Valve (ARV) fails closed.

c. (15) gallons of Boric Acid is added to the Reactor Coolant System.

d. Steam is bypassed around the MSIVs to start the warm-up of the Main Steam Headers.

Answer:

d. Steam is bypassed around the MSIVs to start the warm-up of the Main Steam Headers.

Explanation:

Incorrect	a.	Feedwater is reduced to achieve new S/G level. No change in reactivity.
Incorrect	b.	If an ARV fails shut, RCS temp will go up, and due to a –MTC (middle of cycle) negative reactivity is added causing the point when the Rx will go critical to be later than the ECP.
Incorrect	c.	Adds negative reactivity, causing the point when the Rx will go critical to be later than the ECP.
Correct	d.	If steam is withdrawn, RCS temp will go down, and due to a –MTC (middle of cycle) positive reactivity is added causing the point when the Rx will go critical to be sooner than the ECP.

Examination:	RO 2008 Ginna NRC	**Question #:**	41	Rev. 3	Level: RO
Lesson Plan:	RRT03C	Neutron Life Cycle			
Objective(s):	3.05	Calculate reactivity (or change in reactivity) associated with specific plant conditions (or changing plant conditions) using the reactivity balance equation.			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	039 Main and Reheat Steam K5.08 - Knowledge of the operational implications of the following concepts as the apply to the MRSS: Effect of steam removal on reactivity	**ROI:**	3.6	**SROI:**	3.6
Technical References:	LP RRT03C, Neutron Life Cycle LP RRT05C, Inherent Reactivities	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(1)	55.43			

Question 42

Plant conditions are as follows:

- The unit is operating at 90% power.
- All plant control systems are in their normal at power alignment.

Which of the following failures will cause Reactor Coolant System (RCS) Tavg to **INITIALLY** Rise?

a. Low Pressure Heater Bypass Valve V-3959 fails open.

b. Main Feedwater Bypass Valve to Steam Generator A, AOV-4271, spuriously opens.

c. Instrument Air is lost to "B" Main Feedwater Regulating Valve, AOV-4270.

d. V-5604, Exhaust Steam from the HP Turbine to # 5B FW Heater disc separates from its stem and fails closed.

Answer:

c. Instrument Air is lost to "B" Main Feedwater Regulating Valve, AOV-4270.

Explanation:

Incorrect	a.	Due to the loss of FW heating to both S/Gs, Tavg will lower.
Incorrect	b.	Main Feedwater Bypass Valves are isolated during power ascension. No affect.
Correct	c.	AOV-4270 closes on a loss of IA, FW to the "B" S/G will be isolated. As "B" S/G level drops the heat removal from the RCS drops and Tcold rises. As Tcold rises Tavg rises **INITIALLY**. Following the Rx trip due to "B" S/G level going less than 17%, Tavg will lower.
Incorrect	d.	If ES steam is lost to # 5B FW Heater, FW temp to the B S/G will lower which will cause Thot to lower which will result in Tavg lowering.

Examination:	RO 2008 Ginna NRC	**Question #:**	42	Rev.3	Level: RO
Lesson Plan:	R4301C	Condensate and Feedwater Systems			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Condensate and Feedwater Systems will have on related plant system(s) and/or plant operations. b. Plant operation			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	059 Main Feedwater K3.04 - Knowledge of the effect that a loss or malfunction of the MFW will have on the following: RCS	**ROI:**	3.6	**SROI:**	3.8
Technical References:	LP R4301C, Condensate and Feedwater Systems, Pgs. 56, 57 O-1.2, Plant Startup From Hot Shutdown to Full Load Pgs. 97, 98	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Prairie Island Unit 1 2003 NRC Exam		
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 43

Plant conditions are as follows:

- The unit is operating at 50% power.
- All plant control systems are in their normal at power alignment.
- A leak develops on the Condensate Storage Tanks and cannot be isolated.

As CST water level drops, the **FIRST** volume that will require actions to be taken in accordance with Technical Specifications is when the combined volumes of the CSTs goes below ______ gallons.

a. 25,200

b. 24,350

c. 23,450

d. 22,500

Answer:

b. 24,350

Explanation:

Incorrect	a.	Old ITS requirement prior to up-rate (numbers reversed).
Correct	b.	As per SR 3.7.6.1 24,350 gallons. (ROs take logs on this daily except tank levels read out in %.)
Incorrect	c.	Wrong water volume (numbers reversed from ITS requirements).
Incorrect	d.	Old ITS requirement prior to up-rate.

Examination:	RO 2008 Ginna NRC	**Question #:**	43	Rev. 3	Level: RO
Lesson Plan:	R4201C	Auxiliary Feedwater System			
Objective(s):	1.12	Given a set of plant conditions for the Auxiliary Feedwater Systems, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): b. Describe the basis of any (LCO) and/or Safety Limit			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	061 Auxiliary/Emergency Feedwater 2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications	**ROI:**	3.9	**SROI:**	4.6
Technical References:	ITS 3.7.8	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 44

Plant conditions are as follows:

- The unit is tripped following an accident.
- FR-H.1, Response to Loss of Secondary Heat Sink is being implemented.
- Actions to supply Service Water to the Steam Generators are underway.
- The HCO has attempted to start the "C" SAFW pump, but it will not start.
- The following is the current lineup for the "C" SAFW pump:
    - MOV-9629A, SAFW Pump "C" Suction Valve is closed.
    - AOV-9710A, SAFW Pump "C" Recirculation Valve, is closed.
    - MOV-9703A, SAFW Crossover Valve, is open.
    - MOV-9701A, SAFW Discharge Valve, is closed.

Why won't the "C" SAFW pump start?

a. MOV-9703A, SAFW Crossover Valve must be closed, to prevent runout and damage to the "C" SAFW pump.

b. MOV-9629A, SAFW Pump "C" Suction Valve must be open to provide a suction source to the "C" SAFW pump.

c. AOV-9710A, SAFW Pump "C" Recirculation Valve must be open to ensure the "C" SAFW pump does not run at a shut off head.

d. MOV-9701A, SAFW Discharge Valve must be open, it does not receive any automatic signal to open and is interlocked to not allow the pump to start until MOV-9701A is open.

Answer:

b. MOV-9629A, SAFW Pump "C" Suction Valve must be open to provide a suction source to the "C" SAFW pump.

Explanation:

Incorrect	a.	"C" SAFW pump will start with MOV-9703A open, not a normal alignment but it will not prevent a pump start.
Correct	b.	Interlocked with pump. MOV-9629A must be open to start the "C" SAFW pump.
Incorrect	c.	AOV-9710A, SAFW Pump "C" Recirculation Valve, is normally closed and will automatically open as required while the "C" SAFW pump is running.
Incorrect	d.	MOV-9701A, SAFW Discharge Valve receives any automatic signal to open when the "C" SAFW pump switch is taken to start.

Examination:	RO 2008 Ginna NRC	**Question #:**	44	Rev. 1	Level: RO
Lesson Plan:	R4201C	Auxiliary Feedwater System			
Objective(s):	1.07	Given system conditions, describe the design features of the Auxiliary Feedwater Systems, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. n. Standby Auxiliary Feedwater Pump Motor Operated Discharge Valves. o. Standby Auxiliary Feedwater Pumps p. Standby Auxiliary Feedwater Pump Suction Valves			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	061 Auxiliary/Emergency Feedwater K1.07 - Knowledge of the physical connections and/or cause effect relationships between the AFW and the following systems: Emergency water source	**ROI:**	3.6	**SROI:**	3.8
Technical References:	LP R4201C, Auxiliary Feedwater System pgs. 27, 28 ATT-5.1, Attachment SAFW	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 45

Plant conditions are as follows:

- The unit is at 80% power.
- The electric plant is in a Normal 50/50 lineup.
- Annunciator J-26, Generator Exciter Field Breaker Trip, has just come in.

Which one of the following **DIRECTLY** caused the Generator Exciter Field Breaker to Trip open?

a. Turbine Trip Relay 86X.

b. Turbine Trip Relay 86/BU.

c. Generator Overvoltage (GOV).

d. Overexcitation Protection (OXP).

Answer:

b. Turbine Trip Relay 86/BU.

Explanation:

Incorrect	a.	Trip Relays 86/P and 86/BU trip the Exciter Field Breaker and Trip Relay 86X which in turn prevents the reclosuer of a number of breakers including the Generator Exciter Field Breaker.
Correct	b.	Trip Relays 86/P and 86/BU trip the Exciter Field Breaker.
Incorrect	c.	There are Generator differential current and ground trips, but no Generator Overvoltage trip.
Incorrect	d.	OXP shifts the voltage regulator from auto (AC) to manual (DC), but has no direct trip to the Exciter Field Breaker.

Examination:	RO 2008 Ginna NRC	**Question #:**	45	Rev. 2	Level: RO
Lesson Plan:	R0501C	Main Generation			
Objective(s):	1.07	Given system conditions, describe the design features of the Main Generation system, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Field Breaker			
Category:	Tier 2 /Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	4		
K/A:	062 AC Electrical Distribution K4.02 - Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: Circuit breaker automatic trips	**ROI:**	2.5	**SROI:**	2.7
Technical References:	LP R0501C, Main Generation pgs. 15, 16, 21, 22, 23 AR-J-26, Generator Exciter Field Breaker Trip	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 46

Plant conditions are as follows:

- The unit is at 90% power.
- The electric plant is in a 100/0 lineup.
- Instrument Bus "1C" is being supplied by Inverter "1B".
- The 125 VDC supply from DC distribution panel 1B has been interrupted.

How will the plant respond?

Static Transfer Switch 1B . . .

a. will automatically transfer to the Constant Voltage Transformer, and will automatically transfer back to the "1B" Inverter when 125 VDC is restored.

b. must be manually transferred to the Constant Voltage Transformer, but will automatically transfer back to the "1B" Inverter when 125 VDC is restored.

c. will automatically transfer to the Constant Voltage Transformer, but must be manually transferred back to the "1B" Inverter when 125 VDC is restored.

d. must be manually transferred to the Constant Voltage Transformer, and must be manually transferred back to the "1B" Inverter when 125 VDC is restored.

Answer:

c. will automatically transfer to the Constant Voltage Transformer, but must be manually transferred back to the "1B" Inverter when 125 VDC is restored.

Explanation:

Incorrect	a.	Must be manually transferred back.
Incorrect	b.	Automatically transfers to alt P/S and must be manually transferred back.
Correct	c.	This is the proper operation of the 1B Static Transfer Switch.
Incorrect	d.	Automatically transfers to alt P/S.

Examination:	RO 2008 Ginna NRC	**Question #:**	46	Rev. 2	Level: RO
Lesson Plan:	R0901C	Instrument Bus & DC Power Supply System			
Objective(s):	1.07	Given system conditions, describe the design features of the Instrument Bus & DC Power Supply System			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	062 AC Electrical Distribution A3.05 - Ability to monitor automatic operation of the ac distribution system, including: Safety-related indicators and controls	**ROI:**	3.5	**SROI:**	3.6
Technical References:	LP R0901C, Instrument Bus & DC Power Supply System Pgs. 26, 27	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	C063.0067		
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 47

Plant conditions are as follows:

- The unit is at 100% power.
- The electric plant is in a Normal 50/50 lineup.
- Due to a severe ground there is a complete Loss of "B" DC MCC.

How will the Reactor Trip breakers respond?

a. 1. The "A" Reactor Trip Breaker will be tripped by its shunt coil.
   2. The "B" Reactor Trip Breaker will be tripped by its undervoltage coil.

b. 1. The "A" Reactor Trip Breaker will be tripped by its undervoltage coil.
   2. The "B" Reactor Trip Breaker will be tripped by its shunt coil.

c. 1. The "A" Reactor Trip Breaker will be tripped by its shunt coil.
   2. The "B" Reactor Trip Breaker will be tripped by its shunt coil.

d. 1. The "A" Reactor Trip Breaker will be tripped by its undervoltage coil.
   2. The "B" Reactor Trip Breaker will be tripped by its undervoltage coil.

Answer:

a. 1. The "A" Reactor Trip Breaker will be tripped by its shunt coil.
2. The "B" Reactor Trip Breaker will be tripped by its undervoltage coil.

Explanation:

Correct	a.	Per P-10 and ER-ELEC.2.
Incorrect	b.	Order reversed.
Incorrect	c.	The "B" Reactor Trip Breaker will be tripped by its undervoltage coil.
Incorrect	d.	The "A" Reactor Trip Breaker will be tripped by its shunt coil.

Examination:	RO 2008 Ginna NRC	**Question #:**	47	Rev. 2	Level: RO
Lesson Plan:	R3501C	Reactor Protection System			
Objective(s):	1.03	Describe the physical connections between the Reactor Protection System and a given interrelated system: b. 125 VDC Electrical Distribution System			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	063 DC Electrical Distribution K2.01 - Knowledge of bus power supplies to the following: Major DC loads	**ROI:**	2.9*	**SROI:**	3.1*
Technical References:	ER-ELEC.2, Recovery From Loss of A or B DC Train Att. 1 P-10, Instrument Failure Reference Manual Pgs. 49, 50	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(6)	55.43			

Question 48

Plant conditions are as follows:

- The unit is at 80% power.
- Station Service Transformer – 18 (SST-18) had an internal fault which automatically isolated and deenergized Bus-18.
- The "A" D/G has started and loaded on to Bus-18.
- While checking on the automatic start of the "A" D/G and fault on SST-18, the AO reported MCC "H" is deenergized due to its feeder breaker being open.
- The AO reports an acrid smell in the area of the feeder breaker but no fire.

1. What is the impact of MCC "H" being deenergized?

and

2. What action(s) is/are required to mitigate the consequences of MCC "H" being deenergized?

a. 1. There is no power to the "A" D/G Air Start Compressor.
   2. If a subsequent start is required for "A" D/G, cross connect the "A" and "B" Air Start Systems to enable the "B" Air Start System to start both D/Gs. This crosstie alignment has been considered and no operability concerns exist.

b. 1. There is no power to the "A" D/G Fuel Oil Transfer Pump.
   2. Lineup a temporary pump to fill the "A" D/G Fuel Oil Day Tank. The "A" and "B" D/G Fuel Oil Transfer Systems can not be cross-connected or it will make both D/Gs inoperable.

c. 1. There is no power to the "A" D/G Air Start Compressor.
   2. If a subsequent start is required for "A" D/G, lineup a temporary air compressor to fill the "A" D/G air tank. D/G Air Start Systems can not be cross-connected or it will make both D/Gs inoperable.

d. 1. There is no power to the "A" D/G Fuel Oil Transfer Pump.
   2. Lineup to fill the "A" D/G Fuel Oil Day Tank using the "B" D/G Fuel Oil Transfer Pump. This crosstie alignment has been considered and no operability concerns exist.

Answer:

d. 1. There is no power to the "A" D/G Fuel Oil Transfer Pump.
2. Lineup to fill the "A" D/G Fuel Oil Day tank using the "B" D/G Fuel Oil Transfer Pump. This crosstie alignment has been considered and no operability concerns exist.

Explanation:

Incorrect	a.	There is power to the "A" D/G Air Start Compressor it comes from MCC J.
Incorrect	b.	With MCC H deenergized there is no power to the "A" D/G Fuel Oil Transfer Pump. Temp pump not required, x-conn ops of D/G Fuel Oil Transfer Systems has been considered and no operability concerns exist.
Incorrect	c.	There is power to the "A" D/G Air Start Compressor it comes from MCC J.
Correct	d.	With MCC H deenergized there is no power to the "A" D/G Fuel Oil Transfer Pump. X-conn ops are allowed per T-27.6.

Examination:	RO 2008 Ginna NRC	**Question #:**	48	Rev. 1	Level: RO
Lesson Plan:	R0801C	Diesel Generator System			
Objective(s):	1.04	Describe the (normal cause-effect relationship, functional relationship) between the Diesel Generator System and the given interrelated system: a. 480V electrical System			
	1.05	Identify the electric power supplies to the following system components: a. DG Starting Air Compressors d. Fuel oil system			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	064 Emergency Diesel Generator A2.13 - Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Consequences of opening auxiliary feeder bus (ED/G sub supply)	**ROI:**	2.6*	**SROI:**	2.8*
Technical References:	T-27.6, D/G A/B FOT Pump Isol/Rest pgs. 11,12 P-12, Electrical Systems, Precautions, Limitations, and Setpoints, Att. 1 pg. 11 LP R0801C, Diesel Generator System pg. 26	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 49

Plant conditions are as follows:

- The plant is at 2% power conducting a unit startup per O-1.2, Plant Startup from Hot Shutdown To Full Load.
- "A" and "B" MDAFW pumps are running to maintain S/G water levels at 52%.
- AOV-5738, "A" S/G Blowdown Isolation Valve key switch is in "AUX".
- AOV-5737, "B" S/G Blowdown Isolation Valve key switch is in "AUX".
- S/G Blowdown Valves AOV-5737 / 5738 / 5735 / 5736 "Close – Remote" switch is in "Remote".
- S/G Blowdowns are in progress for both S/Gs to maintain S/G chemistry.
- Chemistry is sampling both S/Gs.

The detector for R-19, Steam Generator Blowdown Process Radiation Monitor fails high.

What is the expected plant condition and plant personnel response for the given conditions? (Assume all ITS, TRM and ODCM requirements are met.)

a. S/G Blowdown valves, AOV-5737 / 5738 are closed, to reopen the S/G Blowdown valves place AOV-5738/5737, A/B S/G Blowdown Isolation Valve key switches to "DEFEAT".

b. S/G Blowdown valves, AOV-5737 / 5738 are open, immediately place S/G Blowdown Valves AOV-5737 / 5738 / 5735 / 5736 "Close – Remote" switch is in "Close".

c. S/G Sample valves, AOV-5735 / 5736 are closed, the only way to reopen the sample valves is to clear the close signal coming from R-19.

d. S/G Sample valves, AOV-5735 / 5736 are open, when the chemist is done sampling and secures the lineup, the S/G Sample valves will close.

Answer:

d. S/G Sample valves, AOV-5735 / 5736 are open, when the chemist is done sampling, the S/G Sample valves will close.

Explanation:

Incorrect	a.	S/G Blowdown valves, AOV-5737 / 5738 will be closed however regardless of the switch position of AOV-5738/7, A/B S/G Blowdown Isolation Valve key switches the S/G Blowdown valves, AOV-5737 / 5738. will not reopen.
Incorrect	b.	S/G Blowdown valves, AOV-5737 / 5738 will be closed regardless of the switch position of AOV-5738/7, A/B S/G Blowdown Isolation Valve key switches.
Incorrect	c.	With the chemist sampling the S/Gs the Hi Rad signal from R-19 is bypassed. When the sample is complete the chemist will place the local Auto/close switch to close which will then close the valve. Clearing the high signal from R-19 is not required to open the sample valves.
Correct	d.	With the chemist sampling the S/Gs the Hi Rad signal from R-19 is bypassed. When the sample is complete the chemist will place the local Auto/close switch to close which will then close the valve.

Examination:	RO 2008 Ginna NRC	**Question #:**	49	Rev. 3	Level: RO
Lesson Plan:	R6601C	Steam Generator Blowdown System			
Objective(s):	1.07	Given system conditions, describe the design features of the Steam Generator Blowdown System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Containment Isolation Valves AOV-5735 and 5736 b. Containment Isolation Valves AOV-5737 and 5738 f. MCB Blowdown Key Defeat Switches (2) g. MCB Blowdown Remote/Close Switch			
	1.10	Predict the effect(s) of a loss or malfunction of the following component(s) and/or instrumentation on the Steam Generator Blowdown System: f. Radiation Monitoring System			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	073 Process Radiation Monitoring A2.02 - Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure	**ROI:**	2.7	**SROI:**	3.2
Technical References:	LP R6601C, Steam Generator Blowdown System pgs. 13,14	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 50

Plant conditions occurred as follows:

- Due to a Loss of Coolant Accident (LOCA) which was followed by significant fuel failure, radiation levels throughout the plant are trending up.
- Control Room (CR) Radiation Monitor alarms were received as follows: (ARM = Area Radiation Monitor and PRM = Process Radiation Monitor)

0800 – R-1, CR ARM Warning Alarm comes in.
0803 – R-45, CR PRM Warning Alarm comes in.
0806 – R-1, CR ARM High Alarm comes in.
0809 – R-46, CR PRM Warning Alarm comes in.
0812 – R-46, CR PRM High Alarm comes in.
0815 – R-45, CR PRM High Alarm comes in.

From the following choices, what is the **FIRST** time that **BOTH** Trains of CREATS system will be in the Emergency mode?

a. 0804

b. 0807

c. 0813

d. 0816

Answer:

c. 0813

Explanation:

Incorrect	a.	Indication only. May choose if unclear on which PRM alarm causes CREATs to go into the Emergency mode.
Incorrect	b.	Indication only. CREATs goes into the Emergency mode on R-45 or 46 not on R-1.
Correct	c.	As per P-9 and AR-E-11. Takes only R45 or R-46 to be in High alarm for both trains of CREATs to go to Emergency mode.
Incorrect	d.	Only takes 1 out of 2 possible RM high alarms from on R-45 or 46. May choose this if uncertain on logic.

Examination:	RO 2008 Ginna NRC	**Question #:**	50	Rev. 1	Level: RO
Lesson Plan:	R3901C	Radiation Monitoring System			
Objective(s):	1.07	Given system conditions, describe the design features of the Radiation Monitoring System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. R-1 i. R-45, R-46			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	073 Process Radiation Monitoring 2.4.45 - Ability to prioritize and interpret the significance of each annunciator or alarm.	**ROI:**	4.1	**SROI:**	4.3
Technical References:	AR-E-11, Cont. Rm. HYVAC Isol. pg. 1 P-9, Radiation Monitor System pgs. 5,13	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(11)	55.43			

Question 51

The UFSAR defines a critical load as one that has either a post-accident function or a function important to nuclear safety.

Which one of the following is **NOT** a critical Service Water load?

a. Residual Heat Removal Pump Room Coolers

b. Containment Recirculation Fan Motor Coolers

c. Preferred Auxiliary Feedwater Pump Oil Coolers

d. Safety Injection Pump Outboard Thrust Bearing Housing Oil Coolers

Answer:

a. Residual Heat Removal Pump Room Coolers

Explanation:

Correct	a.	Non-critical load.
Incorrect	b.	Critical load.
Incorrect	c.	Critical load.
Incorrect	d.	Critical load.

Examination:	RO 2008 Ginna NRC	**Question #:**	51	Rev. 2	Level: RO
Lesson Plan:	R5101C	Service Water System			
Objective(s):	1.02	Given a list of major components or associated instruments of the Service Water System, state their purpose and function.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	076 Service Water 2.1.27 - Knowledge of system purpose and/or function.	**ROI:**	3.9	**SROI:**	4.0
Technical References:	LP R5101C, Service Water System pg. 23	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(8)	55.43			

Question 52

Plant conditions are as follows:

- The unit is at 100% power.
- Due to issues associated with the Service Water (SW) pumps AP-SW.2, Loss of Service Water has been entered.
- "C" SW pump is the only SW Pump running.
- Operations personnel are monitoring plant equipment.

1. What is the expected plant response for the given plant conditions if they remain unchanged?

and

2. What is directed to be performed per AP-SW.2, Loss of Service Water if conditions continue to degrade?

a. 1. Condensate Pump motor temperatures slowly rise, while Condensate Booster Pump motor temperatures remain relatively steady.
   2. Align Alternate D/G Cooling to the "A" D/G.

b. 1. Condensate Pump and Condensate Booster Pump motor temperatures slowly rise.
   2. Align Alternate D/G Cooling to the "A" D/G.

c. 1. Condensate Pump and Condensate Booster Pump motor temperatures slowly rise.
   2. Perform a Load Reduction.

d. 1. Condensate Pump motor temperatures slowly rise, while Condensate Booster Pump motor temperatures remain relatively steady.
   2. Perform a Load Reduction.

Answer:

d. 1. Condensate Pump motor temperatures slowly rise, while Condensate Booster Pump motor temperatures remain relatively steady.
2. Perform a Load Reduction.

Explanation:

Incorrect	a.	Right plant response but, with "C" SW pump as the only SW Pump running, AP-SW.2 directs lining up Alternate D/G Cooling per ER-DG.2 to the "B" D/G.
Incorrect	b.	SW does not supply cooling water to the CB pumps therefore they should stay relatively the same temperature and with "C" SW pump as the only SW Pump running, AP-SW.2 directs lining up Alternate D/G Cooling per ER-DG.2 to the "B" D/G.
Incorrect	c.	SW does not supply cooling water to the CB pumps therefore they should stay relatively the same temperature.
Correct	d.	Right plant response and AP-SW.2 step 5 RNO directs a power reduction to stabilize equipment temperatures.

Examination:	RO 2008 Ginna NRC	**Question #:**	52	Rev. 3	Level: RO
Lesson Plan:	R5101C	Service Water System			
Objective(s):	1.04	Describe the (normal cause-effect relationship, functional relationship) between the Service Water System and the given interrelated system: n. Condensate			
	1.11	Describe indications and/or alarms associated with the Service Water System that would cause entry into an: b. Abnormal Procedure.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	076 Service Water A2.01 - Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of SWS	**ROI:**	3.5*	**SROI:**	3.7*
Technical References:	AP-SW.2, Loss of Service Water pgs. 5,6 LP R5101C, Service Water System pgs. 24	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 53

Plant conditions are as follows:

- The unit is at 100% power.
- The Service and Instrument Air headers are split.
- T-2I, Periodic Operation of Instrument Air Compressors "A" and "B" is being performed.
- "A" and "B" Instrument Air Compressors are running.
- "C" Instrument Air Compressor is running unloaded.
- Instrument Air header pressure then starts to trend down.

Which one of the following is the correct response per T-2I, for the given conditions?

a. "C" Instrument Air Compressor will automatically reload when the Instrument Air header pressure reaches 105 psig.

b. The Control Room Operator must push the "LOAD" button on the MCB (back panel) by the "C" Instrument Air Compressor, when the Instrument Air header pressure reaches 105 psig.

c. Under direction from the Control Room, an Auxiliary Operator (AO) must locally push the "LOAD" button on the "C" Instrument Air Compressor, when the Instrument Air header pressure reaches 105 psig.

d. Under direction from the Control Room, an Auxiliary Operator (AO) must locally push the "LOAD" and "RESET-TEST" buttons on the "C" Instrument Air Compressor, when the Instrument Air header pressure reaches 105 psig.

Answer:

c. Under direction from the Control Room, an Auxiliary Operator (AO) must locally push the "LOAD" button on the "C" Instrument Air Compressor, when the Instrument Air header pressure reaches 105 psig.

Explanation:

Incorrect	a.	No auto reload feature, plausible due to A/B IA Compressors do.
Incorrect	b.	P/B is on local panel.
Correct	c.	Required per T-2I, Notes prior to step 5.3.
Incorrect	d.	Depressing the "RESET-TEST" P/B only required after an emergency shutdown but and not required by T-2I, plausible if uncertain of interlocks/logics.

Examination:	RO 2008 Ginna NRC	**Question #:**	53	Rev. 3	Level: RO
Lesson Plan:	R4701C	Instrument And Service Air System			
Objective(s):	1.07	Given system conditions, describe the design features of the Instrument and Service Air System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes a. Instrument Air Compressors "A" &"B" b. Instrument Air Compressor/ Dryer "C"			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	078 Instrument Air K4.01 - Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control	**ROI:**	2.7	**SROI:**	2.9
Technical References:	T-2I, Periodic Operation of Instrument Air Compressors A and B pg. 2	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 54

Plant conditions are as follows:

- The unit is at 50% power.
- "C" Instrument Air Compressor (IAC) is running.
- "A" and "B" Instrument Air Compressors are in Auto.
- Service and Instrument Air headers are cross-connected per T-1C, Instrument Air/Service Air Cross Connect
- Power to Bus 15 is lost due to a fault.

How will the Control Room Instrument Air (IA) header pressure gauge, PI-2086 respond and why?

When IA header pressure lowers to . . .

a. 105 psig the "A" IAC will automatically start and load. If IA pressure rises to 125 psig, "A" IAC will unload.

b. 105 psig the "B" IAC will automatically start and load. If IA pressure rises to 125 psig, "B" IAC will unload.

c. 105 psig the "A" IAC will automatically start and load. If IA pressure rises to 130 psig, "A" IAC will unload.

d. 105 psig the "B" IAC will automatically start and load. If IA pressure rises to 130 psig, "B" IAC will unload.

Answer:

a. 105 psig the "A" IAC will automatically start and load. If IA pressure rises to 125 psig, "A" IAC will unload.

Explanation:

Correct	a.	With the local switch position in "Auto" the "A" IAC starts on IA header press <105 psig. Constant speed - comp runs loading/unloading to maintain IA header press 110-125 psig.
Incorrect	b.	"B" IAC will not start it is powered from Bus 15, which is dead.
Incorrect	c.	Loads and unloads 110-125 psig.
Incorrect	d.	"B" IAC will not start it is powered from Bus 15, which is dead.

Examination:	RO 2008 Ginna NRC	**Question #:**	54	Rev. 3	Level: RO
Lesson Plan:	R4701C	Instrument and Service Air System			
Objective(s):	1.05	Identify the electric power supplies to the following system components a. Instrument Air Compressors "A" & "B"			
	1.07	Given system conditions, describe the design features of the Instrument and Service Air System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes a. Instrument Air Compressors "A" & "B"			
Category:	Tier 2 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	078 Instrument Air A4.01 - Ability to manually operate and/or monitor in the control room: Pressure gauges	**ROI:**	3.1	**SROI:**	3.1
Technical References:	R4701C, Instrument and Service Air System pg. 15	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 55

Plant conditions are as follows:

- The unit is in Mode 6.
- Shutdown Containment Purge is running to support refueling operations.
- Core Alterations are in progress.
- All pool levels are normal.

Which one of the following would require the suspension of Core Alterations in accordance with Technical Specifications?

a. The Containment Equipment hatch is in place with (6) equally spaced bolts.

b. "A" MCB DC Distribution Panel is isolated to perform a clean and inspect inside the Distribution Panel.

c. The Containment Personnel hatch interlock mechanism is disabled, but one of the two personnel air lock doors remains closed.

d. While the "A" RHR system is in operation the "B" RHR pump motor develops a leak that drains all of the oil out of the motor.

Answer:

b. "A" MCB DC Distribution Panel is isolated to perform a clean and inspect inside the Distribution Panel.

Explanation:

Incorrect	a.	Allowed per ITS Bases 3.9.3. as long as (4) equally spaced bolts are in place the LCO is met.
Correct	b.	With Shutdown Containment Purge running there must be (2) trains of CVI Operable to close the penetrations. With "A" MCB DC Distribution Panel isolated train "A" is inop. LCO NOT met. Must suspend Core Alterations. **This question is in reference to lessons learned at Ginna during a Refueling outage in 2006. DC power was inadvertently tagged out to the CVI isolation ckts. during Refueling operations See CR-2006-005236.**
Incorrect	c.	Allowed per ITS Bases 3.9.3. As long as one of the two personnel air lock doors remains closed, the LCO is met.
Incorrect	d.	Per O-5.1 step 6.1.3 and ITS 3.9.4 with water level >23' above the top of the reactor vessel flange only one RHR pump/Hx must be operable and in operation.

Examination:	RO 2008 Ginna NRC	**Question #:** 55	Rev. 4	Level: RO
Lesson Plan:	R2101C	Containment and Containment Isolation System		
Objective(s):	1.12	Given a set of plant conditions for the Containment and Containment Isolation System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): b. Describe the basis of any (LCO) and/or Safety Limit.		
Category:	Tier 2 / Group 1	**Question Source:**	Bank	
Cognitive level:	C/A	**Difficulty Level:**	3	
K/A:	103 Containment K1.02 - Knowledge of the physical connections and/or cause effect relationships between the containment system and the following systems: Containment isolation/containment integrity	**ROI:** 3.9	**SROI:** 4.1*	
Technical References:	ITS 3.9.3, 3.9.4 and Bases AR-L-4, Safeguard DC Failure CI and CVI Logic	**References provided during Exam:**	None	
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank	
10CFR 55 Content:	55.41(a)(7)	55.43		

Question 56

Which one of the following represents the conditions of the steam entering the PRT from a leaking PORV if pressurizer pressure is 1385 psig and PRT pressure is 5 psig?
(Assume an ideal thermodynamic process)

a. Saturated steam 225-235 °F.

b. Superheated steam 240-250 °F.

c. Superheated steam 260-270 °F.

d. Saturated steam 275-285 °F.

Answer:

c. Superheated steam 260-270 °F.

Explanation:

Incorrect	a.	This is the value if it is determined that the temperature will correspond to the saturation pressure for a PRT pressure of 5 psig plus 15 psi to get 20 psia in the steam tables.
Incorrect	b.	This is the correct value if the line above the 1400 psia line is used on the Mollier Diagram
Correct	c.	If the Mollier Diagram is used to determine the temperature, the applicant could start with the intersection of 1400 psia and the saturation line. A reasonable range of enthalpy at this intersection would be 1172-1175 btu/lbm. Following a constant enthalpy of 1172 btu/lbm to the point where it intersects 20 psia (given PRT pressure of 5 psig plus 15 psia) will yield a temperature almost exactly half-way between 240°F and 280°F. Using the Steam Tables would yield a calculated value of 260.1°F for an enthalpy of 1172 btu/lbm.
Incorrect	d.	If the Mollier Diagram is used to determine the temperature, the applicant could start with the intersection of 1400 psia and the saturation line. A reasonable range of enthalpy at this intersection would be 1172-1175 btu/lbm. If this value is then referenced in the steam table as a saturated gas it will give the temperature as Saturated steam 275-285 °F.

Examination:	RO 2008 Ginna NRC	**Question #:**	56	Rev. 1	Level: RO
Lesson Plan:	R1401C	Pressurizer & Pressurizer Relief Tank			
Objective(s):	1.03	Describe the physical connections between the Pressurizer and Pressurizer Relief Tank System and a given interrelated system a. Reactor Coolant System			
Category:	Tier 2 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	002 Reactor Coolant K1.05 - Knowledge of the physical connections and/or cause-effect relationships between the RCS and the following systems: PRT			**ROI:** 3.2	**SROI:** 3.4
Technical References:	Steam Tables	**References provided during Exam:**	Steam Tables		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Summer Unit 1 2002 NRC Exam		
10CFR 55 Content:	55.41(a)(3)	55.43			

Question 57

Plant conditions are as follows:

- A unit Startup is in progress.
- The Reactor is at 6% power.
- Main Control Board indications show N-35, Intermediate Range Detector, has just failed to 1.0E-3 amps.

How will the plant respond?

a. The reactor has tripped, Source Range indications will automatically become available.

b. The reactor is still at 6% power, N-35 must be bypassed prior to raising power greater than P-10.

c. The reactor has tripped, manual action is required for Source Range indications to become available.

d. The reactor is still at 6% power, as long as N-36 remains Operable, all controls will function normally.

Answer:

c. The reactor has tripped, manual action is required for Source Range indications to become available.

Explanation:

Incorrect	a.	The reactor has tripped due to N-35 failing to the top of the IR scale which is greater than the Trip set point of ~25% Rx power and also satisfies the 1/2 logic for the Rx trip. Manual action will be required because SR detectors will not auto energize because N-35 has failed high and the logic (2/2 IR <5x10-11 amps) won't be met.
Incorrect	b.	The reactor has tripped due to N-35 failing to the top of the IR scale.
Correct	c.	The reactor has tripped due to N-35 failing to the top of the IR scale which is greater than the Trip set point of ~25% Rx power and also satisfies the 1/2 logic for the Rx trip. . Manual action will be required because SR detectors will not auto energize because N-35 has failed high and the logic (2/2 IR <5x10-11 amps) won't be met.
Incorrect	d.	The reactor has tripped due to N-35 failing to the top of the IR scale.

Examination:	RO 2008 Ginna NRC	**Question #:**	57	Rev. 1	Level: RO
Lesson Plan:	R3301C	Nuclear Instrumentation System			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Nuclear Instrumentation System will have on related plant system(s) and/or plant operations. b. Reactor Protection System			
Category:	Tier 2 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	015 Nuclear Instrumentation K3.01 - Knowledge of the effect that a loss or malfunction of the NIS will have on the following: RPS	**ROI:**	3.9	**SROI:**	4.3
Technical References:	LP R3301C ,Nuclear Instrumentation System Pgs. 23,24,38	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(2)	55.43			

Question 58

Plant conditions are as follows:

- The reactor has tripped due to a Loss of Coolant Accident.
- Containment Pressure – 5.5 psig and slowly trending up
- CET Temperature – 380°F and stable
- RCS pressure – 500 psig and stable

Using the provided Fig - 1.0, what is the Subcooling Margin in the RCS using the above parameters?

a. - 80°F Subcooling

b. 0°F Subcooling

c. + 35°F Subcooling

d. + 65°F Subcooling

Answer:

b. 0°F Subcooling

Explanation:

Incorrect	a.	Subcooling Margin derived if 380 psig RCS press and adverse containment values are used.
Correct	b.	Subcooling Margin derived if 500 psig RCS press and adverse containment values are used, which are the correct values to use.
Incorrect	c.	Subcooling Margin derived if 380 psig RCS press and normal containment values are used.
Incorrect	d.	Subcooling Margin derived if 500 psig RCS press and normal containment values are used.

Examination:	RO 2008 Ginna NRC	**Question #:**	58	Rev. 2	Level: RO
Lesson Plan:	REP50C	Emergency Operating Procedures			
Objective(s):	1.01	Define the following terms: a. Adverse Containment			
Category:	Tier 2 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	2		
K/A:	017 In-core Temperature Monitor K5.02 - Knowledge of the operational implications of the following concepts as they apply to the ITM system: Saturation and subcooling of water	**ROI:**	3.7	**SROI:**	4.0
Technical References:	Fig - 1.0, FIGURE MIN SUBCOOLING	**References provided during Exam:**	Fig - 1.0, FIGURE MIN SUBCOOLING		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 59:

Plant conditions are as follows:

- Fuel movement is underway in the Containment and Auxiliary Building.
- Containment Purge is running to support Core Alterations.
- It is discovered that the sample pump that supplies R-10A, Containment Iodine is broken and will not operate.

Which ONE of the following describes how, if at all, fuel movement is affected?

a. No effect. Refueling operations may continue.

b. Refueling operations may only continue as long as R-11 and R-12 are Operable.

c. Immediately suspend all movement of irradiated fuel assemblies in Containment and Core Alterations. Refueling operations may only continue after the sample pump is repaired/replaced and all required testing is completed satisfactorily.

d. Immediately suspend all movement of irradiated fuel assemblies in Containment and Core Alterations. Refueling operations may continue when Refueling integrity is satisfied.

Answer:

d. Immediately suspend all movement of irradiated fuel assemblies in Containment and Core Alterations. Refueling operations may continue when Refueling integrity is satisfied.

Explanation:

Incorrect	a.	The pump that supplies R-10A also supplies R-11 and R-12. R-11 and R-12 are inoperable, refueling activities must be suspended. May chose this answer because R-10A is not a Rad Monitor that will isolate Cnmt Purge.
Incorrect	b.	The pump that supplies R-10A also supplies R-11 and R-12. R-11 and R-12 are inoperable, refueling activities must be suspended. May choose this answer because R-11 and 12 will isolate Cnmt Purge but can't due to broken sample pump.
Incorrect	c.	This is the correct action but if the Cnmt is isolated it is not required to hold up refueling until the sample pump is repaired/replaced. Refueling may continue when refueling integrity is satisfied per O-15.1, O-15.2 and ITS 3.9.3.
Correct	d.	With Containment Purge running and with the sample pump not working (the pump that supplies R-10A also supplies R-11 and R-12) neither R-11 nor R-12 can provide an isolation signal to close the Cnmt Purge valves therefore per ITS 3.3.5 and 3.9.3 all movement of irradiated fuel assemblies in Containment and Core Alterations must be suspended immediately. Refueling may continue when refueling integrity is satisfied per O-15.1, O-15.2 and ITS 3.9.3.

Examination:	RO 2008 Ginna NRC	**Question #:**	59	Rev. 3	Level: RO
Lesson Plan:	R3901C	Radiation Monitoring System			
Objective(s):	1.10	Predict the effect(s) of a loss or malfunction of the following component(s) and/or instrumentation on the Radiation Monitoring System: a. Air sample pump for Rad. Monitor 10A, R-11, R-12 skid.			
	1.12	Given a set of plant conditions for the Radiation Monitoring System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): a. Identify action statements of less than one hour.			
Category:	Tier 2 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	034 Fuel Handling Equipment K6.02 - Knowledge of the effect of a loss or malfunction on the following will have on the Fuel Handling System : Radiation monitoring systems	**ROI:**	2.6	**SROI:**	3.3
Technical References:	LP R3901C, Radiation Monitoring System pg.16 ITS 3.3.5 pgs. 1,2,3,4,5 ITS 3.9.3 pgs. 1,2 O-5.1 Administrative Requirement Checklist or Entry to Mode 6 and Refueling Conditions pg. 21,22	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(11)	55.43			

Question 60

Plant conditions occurred as follows:

- The unit was operating at 100% power.
- Rod control is in Manual.
- All other Control Systems are in their normal alignment.
- One of the Turbine Control Valves disc separated from its stem and failed completely shut.
- During the transient Tavg spiked to 586°F.
- **NO** Operator actions were performed.

Which one of the following describes the proper plant response for these given conditions?

a. All Steam Dumps will snap open and then the Steam Dumps will throttle close until Tavg is approximately equal to Tref.

b. Four Steam Dumps will snap open and then the Steam Dumps will throttle close until Tavg is approximately equal to Tref.

c. Four Steam Dumps will snap open and then the Steam Dumps will throttle close until Tavg is approximately 4°F higher than Tref.

d. All Steam Dumps will snap open and then the Steam Dumps will throttle close until Tavg is approximately 4°F higher than Tref.

Answer:

d. All Steam Dumps will snap open and then the Steam Dumps will throttle close until Tavg is approximately 4°F higher than Tref.

Explanation:

Incorrect	a.	This would be the correct response if Rods were in Auto, with rods in Manual there is a 4°F dead band.
Incorrect	b.	Due to the initial high delta T (>11°F) all the Dump valves will open, with rods in Manual there is a 4°F dead band.
Incorrect	c.	Due to the initial high delta T (>11°F) all the Dump valves will open.
Correct	d.	Due to the initial high delta T (>11°F) all the Dump valves will open, with rods in Manual there is a 4°F dead band.

Examination:	RO 2008 Ginna NRC	**Question #:**	60	Rev. 3	Level: RO
Lesson Plan:	R4501C	Steam Dump System			
Objective(s):	1.07	Given system conditions, describe the design features of the Steam Dump System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. a. Average Tavg.			
Category:	Tier 2 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	041 Steam Dump/Turbine Bypass Control A3.03 - Ability to monitor automatic operation of the SDS, including: Steam flow	**ROI:**	2.7	**SROI:**	2.8
Technical References:	LP R4501C, Steam Dump System pgs. 10,17,21,22	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(4)	55.43			

Question 61

Plant conditions are as follows:

- The unit is at 55% power.
- All control systems are in their normal at power alignment.
- Alarm I-27, Rotor Eccentrity or Vibration Alarm is lit.
- Turbine Bearing Vibrations are as follows:

Bearing	1 - 1.2 mils	4 - 2.7 mils	7 - 5.5 mils
	2 - 1.7 mils	5 - 4.8 mils	8 - 6.0 mils
	3 - 1.0 mils	6 - 5.0 mils	9 - 11.0 mils

The correct **INTIAL** operator response is to:

a. Reduce turbine load to stabilize vibration.

b. Trip the turbine, and perform immediate operator actions for E-0.

c. Trip the turbine, verify power above 1% and verify control rods are driving in to reduce power.

d. Adjust generator hydrogen temp or turbine lube oil temp or exciter cooling to stabilize vibrations.

Answer:

d. Adjust generator hydrogen temperature or turbine lube oil temp or exciter cooling to stabilize vibrations.

Explanation:

Incorrect	a.	This would be the correct response if the high vibrations (>7 mils) were on bearings 1-8.
Incorrect	b.	This would be the correct response per AP-TURB.3, Turbine Vibration RNO step 1 if any turbine vibration was >14 mils since power is > P-9.
Incorrect	c.	This would be the correct response per AP-TURB.3, Turbine Vibration RNO step 1 if any turbine vibration was >14 mils and power was less than 50 % (<P-9).
Correct	d.	This is the correct response per AP-TURB.3, Turbine Vibration RNO step 3.

<table>
<tr><td>Examination:</td><td colspan="2">RO 2008 Ginna NRC</td><td>Question #:</td><td>61</td><td>Rev. 2</td><td>Level: RO</td></tr>
<tr><td>Lesson Plan:</td><td>RAP22C</td><td colspan="5">Turbine Vibration, AP-TURB.3</td></tr>
<tr><td>Objective(s):</td><td>2.01</td><td colspan="5">Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement. (AP-TURB.3).</td></tr>
<tr><td>Category:</td><td colspan="2">Tier 2 / Group 2</td><td>Question Source:</td><td colspan="3">Bank</td></tr>
<tr><td>Cognitive level:</td><td colspan="2">M/F</td><td>Difficulty Level:</td><td colspan="3">4</td></tr>
<tr><td rowspan="2">K/A:</td><td colspan="4" rowspan="2">045 Main Turbine Generator
2.2.44 - Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.</td><td>ROI:</td><td>4.2</td></tr>
<tr><td>SROI:</td><td>4.4</td></tr>
<tr><td>Technical References:</td><td colspan="2">AP-TURB.3, Turbine Vibration pgs. 3,4
ARP I-27, pg. 1</td><td>References provided during Exam:</td><td colspan="3">None</td></tr>
<tr><td>Question History:</td><td colspan="2">Used on Last (2) NRC Exams - No
Used on Audit Exam - No</td><td>Exam Bank #:</td><td colspan="3">Ginna 2004 NRC Exam</td></tr>
<tr><td>10CFR 55 Content:</td><td colspan="2">55.41(a)(10)</td><td colspan="4">55.43</td></tr>
</table>

Question 62

Plant conditions are as follows:

- The unit is at 90% power.
- All systems are in their normal alignment.
- Auxiliary Building (AB) Ventilation IMS Switch is in " FILTER IN".
- "C" Gas Decay Tank (GDT) is being released.

1. Which one of the following is the correct plant response?

and

2. What are the operator actions for that plant response?

a. 1. If R-13, Auxiliary Building Particulate Monitor, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
   2. Verify the GDT release AOV to the plant vent closes.
   Ensure 1F AB Exhaust fan is no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

b. 1. If R-13, Auxiliary Building Particulate Monitor, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
   2. Verify the GDT release AOV to the plant vent closes.
   Ensure 1A, 1B, 1C and 1F AB fans are no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

c. 1. If R-14, Auxiliary Building Gas Monitor, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
   2. Verify the GDT release AOV to the plant vent closes.
   Ensure 1F AB Exhaust fan is no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

d. 1. If R-14, Auxiliary Building Gas Monitor, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
   2. Verify the GDT release AOV to the plant vent closes.
   Ensure 1A, 1B, 1C and 1F AB fans are no longer running.
   Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

Answer:

c. 1. If R-14, Auxiliary Building Gas Monitor, goes into high alarm, "C" Gas Decay Tank release is automatically secured.
2. Verify the GDT release AOV to the plant vent closes.
Ensure 1F AB Exhaust fan is no longer running.
Ensure 1A, 1B and 1C Intermediate Building Exhaust fans are no longer running.

Explanation:

Incorrect	a.	R-13 will cause this realignment of the AB exhaust system but it will not stop the Gas Decay Tank release so RCV-14 (the GDT release AOV to the plant vent) will not close.
Incorrect	b.	R-13 will cause a realignment of the AB exhaust system but it will not stop the Gas Decay Tank release so RCV-14 (the GDT release AOV to the plant vent) will not close. This is correct if the Auxiliary Building (AB) filters are "OUT", but in the stem the Auxiliary Building (AB) filters are "IN", so the additional fans are not affected.
Correct	c.	With a high alarm on R-14 and the AB filters are "IN" the automatic actions that should occur are: RCV-14 (the GDT release AOV to the plant vent) closes. 1F AB Exhaust fan receives a trip signal. 1A, 1B and 1C Intermediate Building fans receive trip signals.
Incorrect	d.	With a high alarm on R-14, RCV-14 (the GDT release AOV to the plant vent) closes. This is correct if the Auxiliary Building (AB) filters are "OUT", but in the stem the Auxiliary Building (AB) filters are "IN", so the additional fans are not affected.

Examination:	RO 2008 Ginna NRC	**Question #:** 62	Rev. 3	Level: RO
Lesson Plan:	R3901C	Radiation Monitoring System		
Objective(s):	1.07	Given system conditions, describe the design features of the Radiation Monitoring System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. c. R-13, R-14		
Category:	Tier 2 / Group 2	**Question Source:**	New	
Cognitive level:	C/A	**Difficulty Level:**	3	
K/A:	071 Waste Gas Disposal A4.09 - Ability to manually operate and/or monitor in the control room: Waste gas release rad monitors		**ROI:** 3.3	**SROI:** 3.5
Technical References:	LP R3901C, Radiation Monitoring System pg. 18 LP R3801C, Waste Disposal System pg. 25	**References provided during Exam:**	None	
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**		
10CFR 55 Content:	55.41(a)(11)	55.43		

Question 63

Plant conditions occurred as follows:

- While operating at rated power, a major steam leak developed downstream of the MSIVs.
- The unit tripped, an automatic Safety Injection (SI) occurred and the MSIVs automatically closed.
- All systems operated as designed.
- RCS pressure has recovered and ARVs are controlling Tavg.
- The Control Room Team is performing E-0, Reactor Trip or Safety Injection.
- A tube rupture then develops in the "B" Steam Generator.

As the Control Room operators monitor their equipment, which Radiation Monitor will indicate the Steam Generator tube rupture first?

a. R-32 – Main Steam Line Radiation Monitor

b. R-47 – Air Ejector/Exhaust Radiation Monitor

c. R-15 – Condenser Air Ejector Radiation Monitor

d. R-19 – Steam Generator Blowdown Radiation Monitor

Answer:

a. R-32 – Main Steam Line Radiation Monitor

Explanation:

Correct	a.	Still available and with flow thru the ARV will be the first RM to indicate the SGTR.
Incorrect	b.	New RM installed last R/F outage to calculate S/G tube leaks however, R-47 was isolated by the MSIVs.
Incorrect	c.	R-15 was isolated by the MSIVs.
Incorrect	d.	R-19 is isolated by the CI that happened when the auto SI occurred.

Examination:	RO 2008 Ginna NRC	**Question #:**	63	Rev. 2	Level: RO
Lesson Plan:	R3901C	Radiation Monitoring System			
Objective(s):	1.02	Given a list of major components or associated instruments of the Radiation Monitoring System, state their purpose and function, and location (when denoted with L): R-19, 15, 32, 47			
Category:	Tier 2 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	072 Area Radiation Monitoring A1.01 - Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ARM system controls including: Radiation levels	**ROI:**	3.4	**SROI:**	3.6
Technical References:	LP R3901C, Radiation Monitoring System pgs. 15,19,20	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(11)	55.43			

Question 64

Plant conditions are as follows:

- The unit is at 100% power.
- "B" and "C" Service Water pumps are running.
- "A" and "D" Service Water pumps are selected.
- The normal supply breaker to Bus 17 fails open.

How will the plant respond?

a. "B" Service Water pump will trip and only "D" Service Water pump will start 40 seconds after the "B" D/G breaker closes.

b. "B" Service Water pump will trip and only "D" Service Water pump will start 17 seconds after the "B" D/G breaker closes.

c. "B" Service Water pump will restart immediately after the "B" D/G breaker closes and "D" Service Water pump will start 17 seconds after the "B" D/G breaker closes.

d. "B" Service Water pump will restart immediately after the "B" D/G breaker closes and "D" Service Water pump will start 40 seconds after the "B" D/G breaker closes.

Answer:

a. "B" Service Water pump will trip and only "D" Service Water pump will start 40 seconds after the "B" D/G breaker closes.

Explanation:

Correct	a.	On an undervoltage the running SW pump will trip and only the selected SW pump will start for that train 40 seconds after "B" D/G breaker closes.
Incorrect	b.	17 seconds is if there was a SI signal concurrent with an undervoltage condition.
Incorrect	c.	There is no immediate restart. There is for CS if it was already running and 17 seconds is if there was a SI signal concurrent with an undervoltage condition.
Incorrect	d.	"B" Service Water pump will trip and not restart because it is not selected.

Examination:	RO 2008 Ginna NRC	**Question #:**	64	Rev. 1	Level: RO
Lesson Plan:	R5101C	Service Water System			
Objective(s):	1.07	Given system conditions, describe the design features of the Service Water System, to include set points, interlocks, and the related automatic actions for the following components, instrumentation and/or processes. d. Safeguards Buses 17 and 18 undervoltage condition.			
Category:	Tier 2 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	075 Circulating Water K2.03 - Knowledge of bus power supplies to the following: Emergency/essential SWS pumps	**ROI:**	2.6*	**SROI:**	2.7*
Technical References:	LP R5101C, Service Water System pgs. 26,27	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(7)	55.43			

Question 65

The following alarms were received simultaneously on the Fire Control Panel in the Control Room:

- Z18 - Control Bldg. 271-0 Relay Room
- First Alarm S08 - Control Bldg. 271-0 Rly Rm Computer Rm Auto Halon

What is the expected condition of the Halon system for the Relay and Mux room one minute after receiving these alarms?

a. Halon system has not actuated.

b. Halon system for Relay and Mux rooms has actuated.

c. Halon system for Relay room has actuated and Mux room has not actuated.

d. Halon system for Relay room has not actuated and Mux room has actuated.

Answer:

a. Halon system has not actuated.

Explanation:

Correct	a.	If the system had actuated, would have the following alarms: First Alarm, Second Alarm and Flow Alarm.
Incorrect	b.	With only the first alarm in the Halon system will not actuate.
Incorrect	c.	With only the first alarm in the Halon system will not actuate.
Incorrect	d.	With only the first alarm in the Halon system will not actuate.

Examination:	RO 2008 Ginna NRC	**Question #:**	65	Rev. 1	Level: RO
Lesson Plan:	R5901C	Fire Protection System			
Objective(s):	1.02	Given a list of major components or associated instruments of the Fire Protection System, state their purpose, function, and location f. Halon Suppression Systems			
Category:	Tier 2 / Group 2	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	086 Fire Protection K4.03 - Knowledge of design feature(s) and/or interlock(s) which provide for the following: Detection and location of fires	**ROI:**	3.1	**SROI:**	3.7
Technical References:	SC-3.16.2.1, Operating Instructions Control Room Fire Panel (FCP) (Gamewell) Panel pgs. 1,2 SC-3.2.7, Immediate Action - Relay/Multiplexer Room Fire pgs. 1,2	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	B086.0003		
10CFR 55 Content:	55.41(a)(8)	55.43			

Question 66

In accordance with CNG-OP-1.01-2001, Communications and Briefings, which of the following is used by Ops Management to communicate short-term information and instructions to Ops personnel and how long is it kept?

a. Standing Order,
30 days maximum.

b. Night Order,
30 days maximum.

c. Standing Order,
Until removed by the GS-Shift Operations or conditions are no longer applicable.

d. Night Order,
Until removed by the GS-Shift Operations or conditions are no longer applicable.

Answer:

d. Night Order,
Until removed by the GS-Shift Operations or conditions are no longer applicable.

Explanation:

Incorrect	a.	Stem states definition of Night Order, Night Orders are maintained as long as they are applicable or until they are removed by the GS-Shift Operations.
Incorrect	b.	Night Orders are maintained as long as they are applicable or until they are removed by the GS-Shift Operations.
Incorrect	c.	Stem states definition of Night Order.
Correct	d.	Per CNG-OP-1.01-2001, Communications and Briefings per section 5.3 B and D.

Examination:	RO 2008 Ginna NRC	**Question #:**	66	Rev. 4	Level: RO
Lesson Plan:	RAD03C	Control Room Conduct			
Objective(s):	1.03	Recognize the duties and responsibilities for the control room personnel.			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.1.15 - Knowledge of administrative requirements for temporary management directives, such as standing orders, night orders, Operations memos, etc.	**ROI:**	2.7	**SROI:**	3.4
Technical References:	CNG-OP-1.01-2001, Communications and Briefings per section 5.3 pgs. 6,7 Copy of a real Night Order	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 67

During the implementation of E-0, Reactor Trip or Safety Injection, which one of the following communications is correct, in accordance with CNG-HU-1.01-1001, Human Performance Tools and Verification Practices?

a. CRS: Bob.
Bob: Ready.
CRS: Start an Aux Feed Water Pump.
Bob: Start an Aux Feed Water Pump.
CRS: That is correct.

b. CRS: Bob, start an Aux Feed Water Pump.
Bob: Start an Aux Feed Water Pump.
CRS: That is correct.

c. CRS: Bob.
Bob makes eye contact with the CRS.
CRS: Start "Bravo" Aux Feed Water Pump.
Bob: Start an Aux Feed Water Pump.
CRS: That is wrong. Start "Bravo" Aux Feed Water Pump.
Bob: Start "Bravo" Aux Feed Water Pump.
CRS: That is correct.

d. CRS: Bob, start "Bravo" Aux Feed Water Pump.
Bob: Start an Aux Feed Water Pump.
CRS: That is incorrect. Start "Bravo" Aux Feed Water Pump.
Bob: Start "Bravo" Aux Feed Water Pump.
CRS: That is correct.

Answer:

c. CRS: Bob.
Bob makes eye contact with the CRS.
CRS: Start "Bravo" Aux Feed Water Pump.
Bob: Start an Aux Feed Water Pump.
CRS: That is wrong. Start "Bravo" Aux Feed Water Pump.
Bob: Start "Bravo" Aux Feed Water Pump.
CRS: That is correct.

Explanation:

Incorrect	a.	Does not specify which AFW pump to start. Order should not be put out this way and receiver should not allow order to continue without clarification.
Incorrect	b.	The CRS should get Bobs attention first then deliver his message. Also the message does not specify which AFW pump to start. Order should not be put out this way and receiver should not allow order to continue without clarification.
Correct	c.	CRS identifies message was repeated incorrectly and says "wrong" as per CNG-HU-1.01-1001 section 5.7.B.3.e then redelivers the message and ensures it is understood correctly.
Incorrect	d.	The CRS should get Bobs attention first then deliver his message. Per CNG-HU-1.01-1001 section 5.7.B.3.e, must use "wrong not "that is incorrect".

Examination:	RO 2008 Ginna NRC	**Question #:**	67	Rev. 3	Level: RO
Lesson Plan:	RAD03C	Control Room Conduct			
Objective(s):	3.02	State the general considerations for all personnel performing work in the control room.			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	2.1.38 - Knowledge of the station's requirements for verbal communications when implementing procedures.	**ROI:**	3.7*	**SROI:**	3.8
Technical References:	CNG-HU-1.01-1001, Human Performance Tools and Verification Practices Pgs. 21,22	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 68

Which ONE (1) of the following describes the **MINIMUM** Source Range Nuclear Instrumentation requirement that must be met prior to off-loading fuel from the reactor vessel?

	Visual in Control Room	Audible in Control Room	Audible in Containment
a.	1	1	0
b.	1	1	1
c.	2	1	0
d.	2	0	1

Answer:

c.	2	1	0

Explanation:

Incorrect	a.	2/1/0 is the ITS requirements. May choose if its believed as long as any (1) SR is operable and working in the control room area the Refueling requirements are met.
Incorrect	b.	2/1/0 is the ITS requirements. May choose if its believed as long as any (1) SR is operable and working in all the areas the Refueling requirements are met.
Correct	c.	This is correct per ITS 3.9.2 and O-15.1.
Incorrect	d.	2/1/0 is the ITS requirements. May choose if its believed that (2) SR are required to be operable and working in the control room area and only 1 working in the Cnmt.

Examination:	RO 2008 Ginna NRC	**Question #:**	68	Rev. 4	Level: RO
Lesson Plan:	R3301C	Nuclear Instrumentation System			
Objective(s):	1.12	Given a set of plant conditions for the Nuclear Instrumentation System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): a. Identify action statements of less than one hour.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.1.40 - Knowledge of refueling administrative requirements.	**ROI:** 2.8	**SROI:** 3.9		
Technical References:	ITS 3.9.2 pgs. 1,2 ITS 3.9.2 Bases pg. 1 O-5.1 Administrative Requirement Checklist or Entry to Mode 6 and Refueling Conditions pg. 11	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 69

Which one of the following jobs would **REQUIRE** an Isolated Work Area (a HOLD), due to it meeting the definition of a "Hazardous Energy" in accordance with A-1401, Station Holding Rules? (Assume **NO** instrument calibrations will be performed.)

Work on a(n) . . .

a. AC circuit where the maximum voltage is 40 volts AC.

b. DC circuit where the maximum voltage is 55 volts DC.

c. system which has a maximum pressure of 30 psig and a maximum temperature of 90°F.

d. system which has a maximum pressure of 15 psig and a maximum temperature of 100°F.

Answer:

b. DC circuit where the maximum voltage is 55 volts DC.

Explanation:

Incorrect	a.	Per A-1401 a hold is required for > 50 volts (AC OR DC).
Correct	b.	Per A-1401 a hold is required for > 50 volts (AC OR DC).
Incorrect	c.	Per A-1401 a hold is required for > 50 psig hydraulic pressure and/or > 120°F temperature.
Incorrect	d.	Per A-1401 a hold is required for > 50 psig hydraulic pressure and/or > 120°F temperature.

Examination:	RO 2008 Ginna NRC	**Question #:**	69	Rev. 2	Level: RO
Lesson Plan:	RAD30C	Station Holding Rules			
Objective(s):	1.02	Define the following common terms: f. Isolated Work Area			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.2.13 - Knowledge of tagging and clearance procedures.	**ROI:**	4.1	**SROI:**	4.3
Technical References:	A-1401, Station Holding Rules pg. 6	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 70

Which one of the following plant conditions would require entry into a Technical Specification Action Statement?

a. While in Mode 5, the HCO reports the Refueling Water Storage Tank (RWST) borated water volume is 275,000 gallons.

b. Chemistry reports, Spent Fuel Pool boron concentration is 2400 ppm and there are fuel assemblies in the Spent Fuel Pool.

c. During the roll up of the Turbine Generator, the CO reports "A" Loop Tavg is 545°F, "B" Loop Tavg is 535°F and both are trending up slowly.

d. During a Reactor Startup, just after the Shutdown Bank rods are pulled to step 223, it is discovered that one of the Shutdown Rods is untrippable.

Answer:

c. During the roll up of the Turbine Generator, the CO reports "A" Loop Tavg is 545°F, "B" Loop Tavg is 535°F and both are trending up slowly.

Explanation:

Incorrect	a.	RWST level is not applicable in Mode 5 per ITS 3.5.4. (Less than a one hour LCO.)
Incorrect	b.	Spent Fuel Pool boron concentration minimum limit is 2300 ppm per ITS 3.7.12. (Less than a one hour LCO.)
Correct	c.	When in Mode 1, **BOTH** RCS Tavgs must be > 540°F per ITS 3.4.2. (Less than a one hour LCO.)
Incorrect	d.	Still in Mode 3 per O-1.2, Plant Startup from Hot Shutdown to Full Load, therefore LCO 3.1.4 is not applicable. (Less than a one hour LCO.)

Examination:	RO 2008 Ginna NRC	**Question #:**	70	Rev. 2	Level: RO
Lesson Plan:	R1001C	Reactor Coolant System			
Objective(s):	1.12	Given a set of plant conditions for the Reactor Coolant System, perform the following in accordance with Technical Specifications, TS Bases, TRM: a. Identify action statements of less than one hour.			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	2.2.42 - Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	**ROI:**	3.9	**SROI:**	4.6
Technical References:	ITS: 3.5.4, 3.7.12, 3.4.2 3.1.4, Table 1.1-1 O-1.2, Plant Startup from Hot Shutdown to Full Load pg. 18,19	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 71

Plant conditions occurred as follows:

- The plant is operating at 90% power.
- The HCO notes that R-29, Containment High Range monitor indication, rapidly rises to almost full scale and then it lowers back to zero.
- One minute later R-30, Containment High Range monitor does the same thing.
- No Control Room alarms came in during this period.
- The HCO reports this to the CRS.

Which ONE of the below is correct for this indication?

a. Write a Condition Report on R-29 and R-30 and declare them Inoperable per Technical Specifications.

b. Initiate S-12.2, Operator Action in the Event of Significant Increase in Leakage.

c. This is an expected indication, it's an automatic electronic source check.

d. Enter AP-RCS.1, Reactor Coolant Leak.

Answer:

c. No action required, this is an automatic electronic source check.

Explanation:

Incorrect	a.	Correct response for failed equipment, not correct for R-29 and R-30 as this is a normal systems check it does every 17.1 minutes.
Incorrect	b.	High Rads in the containment may lead operators to look for leaks; however, since there are no other indications of leakage entry into S-12.2 would not be required.
Correct	c.	Per the first note and caution in S-14.3, Operation of Containment High Range Area Monitors R-29, R-30, WHEN the monitor is operating normally, THEN the system performs an Electronic Check Source (ECS) test every 17.1 minutes. and during the ECS test, all alarms are muted; that is, their operation is disabled until the completion of the test. During this six-second period there is no warning of a high-radiation condition.
Incorrect	d.	Possible indication of a leak in the RCS but with no other indications entry into AP-RCS.1 is not required.

Examination:	RO 2008 Ginna NRC	**Question #:**	71	Rev. 3	Level: RO
Lesson Plan:	R3901C	Radiation Monitoring System			
Objective(s):	1.09	Given a specific fundamental principle, performance test, limits and/or precautions for the Radiation Monitoring System or component, describe the reason/basis and the effect on the operations of the system or component.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.3.5 - Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personal monitoring equipment, etc.	**ROI:**	2.9	**SROI:**	2.9
Technical References:	LP R3901C, Radiation Monitoring System S-14.3, Operation of Containment High Range Area Monitors R-29, R-30 pg. 6	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	B072.0001		
10CFR 55 Content:	55.41(a)(11)	55.43			

Question 72

Plant conditions are as follows:

- The Reactor is at 2% power.
- A containment entry is underway to look for a leak identified during the startup.
- The mechanic and contractor RP tech who are looking for the leak, have called the Control Room and informed you they have found a Boric Acid leak coming from the reactor cavity area and going into the "A" sump.
- They want permission to enter the reactor cavity area and "A" sump to determine the extent of the Boric Acid leak.
- The contractor RP tech says that he has a portable radiation monitoring device with him that will cover the expected doses, and they are both wearing all the required dosimitry.

What is the correct response in accordance with A-3, Containment Vessel Access Requirements?

a. They may not enter either of the areas at this time.

b. They may enter the "A" sump, but not the reactor cavity area.

c. They may enter the reactor cavity area, but not the "A" sump.

d. As long as the RP tech stays with the mechanic, they can enter both areas.

Answer:

a. They may not enter either of the areas at this time.

Explanation:

Correct	a.	Per A-3, Containment Vessel Access Requirements precaution 3.5: When the reactor is critical, personnel SHALL NOT enter the reactor cavity or A Sump.
Incorrect	b.	Entering sump might be possible with RP support however procedurally not allowed.
Incorrect	c.	Entering reactor cavity area might be possible with RP support however procedurally not allowed.
Incorrect	d.	With RP coverage, lends validity to the possibility of entering either area however procedurally not allowed.

Examination:	RO 2008 Ginna NRC	**Question #:**	72	Rev. 2	Level: RO
Lesson Plan:	RAD02C	Containment Vessel Access Requirements			
Objective(s):	1.02	Identify the restrictions concerning entries into CNMT, or restrictions to operational activities with personnel inside CNMT.			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.3.12 - Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc	**ROI:**	3.2	**SROI:**	3.7
Technical References:	A-3, Containment Vessel Access Requirements Pg. 3	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(12)	55.43			

Question 73

Plant conditions are as follows:

- The Control Room staff is responding to an Orange path on Containment per FR-Z.1, Response To High Containment Pressure.
- The STA provides an update with the following information:
  - o Inventory - Yellow
  - o Containment - Red
  - o Heat Sink - Red
  - o Subcritical - Green
  - o Integrity - Orange
  - o Core Cooling - Red

What is the correct action to be taken and why?

a. Immediately transition to FR-C.1, Response To Inadequate Core Cooling because efforts need to be made to ensure the fuel cladding remains intact.

b. Immediately transition to FR-H.1, Response To Loss of Heat Secondary Sink, because efforts need to be made to ensure the Reactor Coolant System remains intact.

c. Complete FR-Z.1, Response To High Containment Pressure, because the Orange path is now a Red path and efforts need to be made to ensure Containment, the last barrier to the public, remains intact, then transition to FR-C.1, Response To Inadequate Core Cooling.

d. Complete FR-Z.1, Response To High Containment Pressure, because the Orange path is now a Red path and efforts need to be made to ensure Containment, the last barrier to the public, remains intact, then transition to FR-H.1, Response To Loss of Secondary Heat Sink.

Answer:

a. Immediately transition to FR-C.1, Response To Inadequate Core Cooling because efforts need to be made to ensure the fuel cladding remains intact.

Explanation:

Correct	a.	Even with a Cnmt Red path, the Red path for Core Cooling is a higher priority and must be performed first.
Incorrect	b.	Correct to leave Cnmt Orange/ Red path, but Core Cooling is a higher priority then Loss of Secondary Heat Sink.
Incorrect	c.	Required to immediately exit FR-Z.1 and enter FR-C.1 due to it being the highest Red path at this time.
Incorrect	d.	Required to immediately exit FR-Z.1 and enter FR-C.1 due to it being the highest Red path at this time.

Examination:	RO 2008 Ginna NRC	**Question #:**	73	Rev. 2	Level: RO
Lesson Plan:	RFR00C	Emergency Operating Procedures			
Objective(s):	1.01	List the Critical Safety Function Status Trees (CSFSTs) in order of priority.			
	1.03	State the purpose of the Critical Safety Function (CSFs) and the Critical Safety Function Status Trees (CSFSTs).			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	2.4.22 - Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.	**ROI:**	3.6	**SROI:**	4.4
Technical References:	LP RFR00C, Emergency Operating Procedures Pgs. 7,8,10,11,1213,14	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 74

Plant conditions are as follows:

- The plant is in a Site Area Emergency.
- Due to degrading plant conditions an entry into containment must be made in the next 15 minutes to prevent a severe radiation release to the public from occurring and escalating the site to a General Emergency.
- RP has determined the volunteer will receive 3500 mrem (TEDE) for the evolution.

What is the **MINIMUM** authorization required to approve the expected exposure in accordance with EPIP 2-8, Voluntary Acceptance of Emergency Radiation Exposure?
(Answers listed are from lowest to highest authorization required.)

a. Radiation Protection Technician

b. Supervisor, Radiation Protection Operations

c. Radiation Protection Manager

d. Emergency Coordinator

Answer:

c. Radiation Protection Manager

Explanation:

Incorrect	a.	Radiation Protection Manager authorization is required. Possible choice due to time issue.
Incorrect	b.	Radiation Protection Manager authorization is required. Possible choice due to time issue.
Correct	c.	Required per EPIP 2-8, Voluntary Acceptance of Emergency Radiation Exposure pg. 2. Radiation Protection Manager can authorize up to (4) rem (TEDE).
Incorrect	d.	Emergency Coordinator approval required for >4 rem (TEDE).

Examination:	RO 2008 Ginna NRC	**Question #:**	74	Rev. 1	Level: RO
Lesson Plan:	RSC01C	Emergency Response Organization & Responsibilities			
Objective(s):	6.00	State the function and responsibilities for Operations personnel and for the RP technician for any emergency event classification level.			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.4.37 - Knowledge of the lines of authority during implementation of the emergency plan.	**ROI:**	3.0	**SROI:**	4.1
Technical References:	EPIP 2-8, Voluntary Acceptance of Emergency Radiation Exposure pg. 2	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41(a)(12)	55.43			

Question 75

Plant conditions are as follows:

- The plant is in a General Emergency.
- The Ginna Station Nuclear Emergency Response Plan is being implemented.

From the list below, which one of the following correctly identifies the responsibilities of the HCO/CO during the General Emergency in accordance with EPIP 5-7, Emergency Organization?

1. Report all unusual observations or communications to the Shift Manager.
2. Sound the alarm and make announcements as necessary.
3. Advise the Shift Manager in the diagnosis of unusual event conditions and above.
4. Check that the Control Room ventilation system is in re-circulation mode.
5. Assist as directed and inform Shift Manager of all Control Room changes.
6. Ensure search and rescue is initiated, if necessary, per EPIP 1-8.

a. 1, 3, 5

b. 2, 4, 6

c. 1, 3, 6

d. 2, 4, 5

Answer:

d. 2, 4, 5

Explanation:

Incorrect	a.	# 1 is the responsibility of the C/R Communicator. # 3 is the responsibility of the STA per EPIP 5-7, Emergency Organization.
Incorrect	b.	# 6 is the responsibility of the Emergency Coordinator per EPIP 5-7, Emergency Organization.
Incorrect	c.	# 1 is the responsibility of the C/R Communicator. # 3 is the responsibility of the STA per EPIP 5-7, Emergency Organization.. # 6 is the responsibility of the Emergency Coordinator.
Correct	d.	Per EPIP 5-7, Emergency Organization pg 33.

Examination:	RO 2008 Ginna NRC	**Question #:**	75	Rev. 3	Level: RO
Lesson Plan:	RSC01C	Emergency Response Organization & Responsibilities			
Objective(s):	6.0	State the function and responsibilities for Operations personnel and for the RP technician for any emergency event classification level.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.4.39 - Knowledge of RO responsibilities in emergency plan implementation.	**ROI:**	3.9	**SROI:**	3.8
Technical References:	EPIP 5-7, Emergency Organization pg. 26,29,30,33	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41(a)(10)	55.43			

Question 76

Plant conditions occurred at the following times:

1330 - While operating at rated power, the following alarms came in:

- A-6, CONT Spray Pump Cooling Water Out Lo Flow
- A-7(15), RCP CCW Return Hi Temp or Lo Flow
- A-9, RHR Pump Cooling Water Outlet Lo Flow
- A-14, Safety Inj Pumps Cooling Water Out Lo Flow
- A-17, Motor Off RCP CCWP
- A-22, CCW Pump Discharge Lo Pressure

1331 - Standby CCW pump trips when it auto-starts leaving no CCW pumps running.
1332 - Surge Tank level is trending up.
1333 - R-17, CCW Radiation Monitor is in alarm and counts are trending up slowly.

Based on the information given, which of the following is/are the correct action(s) to be taken **FIRST** at 1333?

a. Direct entry into AP-CCW.1, Leak into Component Cooling Loop and verify that the CCW surge tank vent valve has automatically isolated.

b. Direct entry into AP-CCW.2, Loss of CCW During Power Operation and trip the reactor, perform the Immediate actions of E-0, and then trip the RCPs.

c. Direct entry into AP-CCW.2, Loss of CCW During Power Operation and if a RCP bearing temp exceeds 150 degrees F, trip the reactor and then the RCPs.

d. Direct entry into AP-CCW.1, Leak into Component Cooling Loop and verify seal injection flow normal, then manually close the thermal barrier return valves (AOV-754A/754B).

Answer:

b. Direct entry into AP-CCW.2, Loss of CCW during Power Operation and trip the reactor, perform the Immediate actions of E-0, and then trip the RCPs.

Explanation:

Incorrect	a.	Correct expected action, but with no CCW pumps running, tripping the unit and securing the RCPs is the priority.
Correct	b.	Most important item to perform per AP-CCW.2 step 1a RNO, with no CCW pumps running tripping the unit and RCPs are the priority.
Incorrect	c.	With no CCW pumps running per the AP, are to trip the unit and securing the running RCPs, this choice is not correct because the AP directs tripping only the RCP that exceeds 200 degrees F not both RCPs. 150 degrees F is an alarm setpoint for high seal water supply temperature to the RCPs (A-8 and A-16).
Incorrect	d.	Correct for AP-CCW.1, but with no CCW pumps running tripping the unit and securing the RCPs is the priority.

Examination:	RO 2008 Ginna NRC	**Question #:**	76	Rev. 4	Level: SRO
Lesson Plan:	RAP02C	Loss Of CCW During Power Operations, AP-CCW.2			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in AP-CCW.2, Loss of CCW During Power Operation.			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000015/000017 RCP Malfunctions AA2.10 - Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to secure RCPs on loss of cooling or seal injection	**ROI:**	3.7	**SROI:**	3.7
Technical References:	AP-CCW.1, Leak into Component Cooling pg. 3 AP-CCW.2, Loss of CCW during Power Operation pg. 4	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 77

The control room has just been informed by the system engineer that non-quality parts were used to rebuild both Component Cooling Water (CCW) pumps during the last outage. The CCW system has passed all of its surveillances and has been operating satisfactorily for the past 6 months. It was discovered these parts were sent by mistake with the correct part numbers but were made of substandard material.

Which one of the following applies?

a. No actions required, both CCW pumps are "Operable". The CCW pumps have passed all surveillances and testing.

b. Declare both CCW pumps "Operable but Degraded". Immediately initiate action for Engineering to perform an "Operability Determination".

c. Declare both CCW pumps "Inoperable". Immediately enter LCO 3.0.3 due to no other Tech Spec applies and perform shutdown when allowed by Tech Specs.

d. Declare both CCW pumps "Inoperable". Immediately initiate action to restore one CCW train to Operable status within (1) hour. If action not completed after (1) hour be in Mode 3 in (6) hours and Mode 4 in (12) hours when allowed by Tech Specs.

Answer:

b. Declare both CCW pumps "Operable but Degraded". Immediately initiate action for Engineering to perform an "Operability Determination".

Explanation:

Incorrect	a.	At a minimum a safety determination should be done, not "no action required". Even though it passed all its testing and surveillances it is still degraded due to the wrong parts being installed.
Correct	b.	Engineering would perform an Operability Determination to verify the pumps could perform their safety functions for and at the times required by ITS and per the FSAR. If it passes the Operability Determination, then additional engineering evaluations would be performed on the parts. If it does not pass the Operability Determination, it would then be declared Inop.
Incorrect	c.	There is an ITS for both CCW trains inop. Entry into LCO 3.0.3 is not required.
Incorrect	d.	ITS 3.7.7 does not allow (1) hour to restore a train.

Examination:	RO 2008 Ginna NRC	**Question #:**	77	Rev. 4	Level: SRO
Lesson Plan:	R2801C	Component Cooling Water System			
Objective(s):	1.12	Given a set of plant conditions for the Component Cooling Water System, perform the following in accordance with (Technical Specifications, TS Bases). a. Identify action statements of less than one hour			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	4		
K/A:	000026 Loss of Component Cooling Water 2.2.37 - Ability to determine operability and/or availability of safety related equipment.	**ROI:**	3.6	**SROI:**	4.6
Technical References:	CNG-OP-1.01-1002, Conduct Of Operability Determinations/Functionality Assessments pg. 5 ITS 3.7.7 condition D pgs. 3.7.7-1 and 2	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(2)			

Question 78

Plant conditions occurred as follows:

- The unit was operating at 20% power.
- An RCS leak has caused Pressurizer pressure to lower to 1850 psig.
- Power remains near 10% and both Reactor trip breakers still indicate closed.
- The operators are performing Step 1 of E-0, Reactor Trip or Safety Injection.
- Pressurizer level is 10% and stable.

Which ONE of the following describes the correct direction to give the crew?

a. Continue with E-0 and manually initiate SI.

b. Continue with E-0 and **DO NOT** manually initiate SI.

c. Transition to FR-S.1, Response to Nuclear Power Generation/ATWS and manually initiate SI.

d. Transition to FR-S.1, Response to Nuclear Power Generation/ATWS and **DO NOT** manually initiate SI.

Answer:

d. Transition to FR-S.1, Response to Nuclear Power Generation/ATWS and **DO NOT** manually initiate SI.

Explanation:

Incorrect	a.	E-0 steps/actions no longer apply, a crew transition to FR-S.1 is required to mitigate the ATWS. May chose to perform this action based on power dropping from 20% to 10 % and Pzr pressure being within 100 psig of the SI set point.
Incorrect	b.	The crew must immediately transition to FR-S.1 to mitigate the ATWS, E-0 no longer applies while in FR-S.1 May chose to perform this action based on power dropping from 20% to 10 % and Pzr pressure still being greater than 100 psig above the SI set point and Pzr level is stable.
Incorrect	c.	FR-S.1 has no instructions/RNO for manually initiating SI. May chose to perform this action based on Pzr pressure being within 100 psig of the SI set point.
Correct	d.	Must transition per E-0 step 1d RNO. No guidance is given in any RNO or procedure step to initial SI, only to verify SI equipment alignment if running.

Examination:	RO 2008 Ginna NRC	**Question #:**	78	Rev. 3	Level: SRO
Lesson Plan:	REP00C	Reactor Trip or Safety Injection E-0			
Objective(s):	1.03	State the immediate actions of E-0, Reactor Trip or Safety Injection.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000029 ATWS 2.1.23 - Ability to perform specific system and integrated plant procedures during all modes of plant operation.	**ROI:**	4.3	**SROI:**	4.4
Technical References:	FR-S.1, Response to Reactor Restart/ATWS E-0, Reactor Trip or Safety Injection pg. 3	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Turkey Point Unit 3 2003 NRC Exam		
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 79

Plant conditions occurred as follows

- The plant was operating at rated power when the unit experienced a fire in the cable tunnel.
- ER-FIRE.0, CR Response to Fire Alarms and Reports is being performed.
- Just before the fire was reported as being extinguished there were numerous alarms, the reactor tripped and there was a complete loss of power to the "B" DC Train.

What procedure is entered to restore power to the "B" DC system and how will it be accomplished?

a. ER-ELEC.2, Recovery from Loss of A or B DC Train. The TSC Battery will be cross tied to the "B" DC Train and the TSC Diesel will supply the TSC Battery Chargers.

b. ER-FIRE.2, Alternate Shutdown for Cable Tunnel Fire. The "B" DC Train Battery chargers will be back fed from the TSC Diesel through Bus 15 and the Bus 15-16 cross-tie.

c. ER-ELEC.2, Recovery from Loss of A or B DC Train. The "B" DC Train Battery chargers will be back fed from the TSC Diesel through Bus 15 and the Bus 15-16 cross-tie.

d. ER-FIRE.2, Alternate Shutdown for Cable Tunnel Fire. The TSC Battery will be cross tied to the "B" DC Train and the TSC Diesel will supply the TSC Battery Chargers.

Answer:

d. ER-FIRE.2, Alternate Shutdown for Cable Tunnel Fire. The TSC Battery will be cross tied to the "B" DC Train and the TSC Diesel will supply the TSC Battery Chargers.

Explanation:

Incorrect	a.	ER-ELEC.2, Recovery from Loss of A or B DC Train deals with the loss of the bus and how it affects the plant, with a known fire in the cable tunnel area ER-FIRE.2 should be entered. ER-ELEC.2 gives no specific procedure steps to restore power after electricians have investigated the problem. The actions are correct per ER-FIRE.2.
Incorrect	b.	ER-FIRE.2, Alternate Shutdown for Cable Tunnel Fire provides guidance in Att. 8 section 2, to provide long term DC power. The actions given are plausible to restore power but not called out for per ER-FIRE.2.
Incorrect	c.	ER-ELEC.2, Recovery from Loss of A or B DC Train deals with the loss of the bus and how it affects the plant, with a known fire in the cable tunnel area ER-FIRE.2 should be entered. ER-ELEC.2 gives no specific procedure steps to restore power after electricians have investigated the problem. The actions given are plausible to restore power but not called out for per ER-FIRE.2.
Correct	d.	ER-FIRE.2, Alternate Shutdown for Cable Tunnel Fire provides guidance in Att. 8 section 2, to provide long term DC power. The actions are correct per ER-FIRE.2.

Examination:	RO 2008 Ginna NRC	**Question #:**	79	Rev. 3	Level: SRO
Lesson Plan:	RSC62D	Appendix R - Alternative Shutdown			
Objective(s):	RSC62D6	Demonstrate that appropriate equipment and procedures are used to mitigate the fire.			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000058 Loss of DC Power 2.1.20 - Ability to interpret and execute procedure steps.	**ROI:**	4.6	**SROI:**	4.6
Technical References:	ER-FIRE.2, Alternate Shutdown for Cable Tunnel Fire pgs. 3,30-35 ER-FIRE.0, CR Response to Fire Alarms & Reports pg. 4 ER-ELEC.2, Recovery from Loss of A or B DC Train	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 80

Plant conditions occurred as follows:

- A Reactor Coolant Leak has been identified in the Auxiliary Building.
- Efforts to enter the area to isolate the leak have been unsuccessful.
- The Control Room team has worked their way through the EOPs and are about to leave ECA-1.2, LOCA Outside Containment.
- RWST level is 55% and lowering slowly.
- Reactor Coolant System pressure and Pressurizer water level continue to lower slowly.

Which procedure will be transitioned to and why?

a. E-1, Loss of Reactor or Secondary Coolant, to continue actions to address the LOCA.

b. ECA-1.1, Loss of Emergency Coolant Recirculation, to identify the location of the LOCA and isolate it.

c. E-1, Loss of Reactor or Secondary Coolant, to initiate actions to lower RCS pressure to minimize break flow to slow the loss of inventory.

d. ECA-1.1, Loss of Emergency Coolant Recirculation, to address the loss of inventory available for core cooling.

Answer:

d. ECA-1.1, Loss of Emergency Coolant Recirculation, to address the loss of inventory available for core cooling.

Explanation:

Incorrect	a.	Per step 7 of ECA-1.2, only if the leak is isolated you are to return to E-1. Per the background document: The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions.
Incorrect	b.	Per step 7 RNO, of ECA-1.2, if the leak is NOT isolated you are to go to ECA-1.1 but while it is prudent to continue to look for the leak, ECA-1.1 is going to address the loss of inventory, to make the RWST last as long as possible by minimizing its depletion and makeup to the RWST.
Incorrect	c.	Per step 7 of ECA-1.2, only if the leak is isolated you are to return to E-1. Per the background document: The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. The action to lower RCS pressure to slow the leak rate would to be prudent but not the stated reason.
Correct	d.	Per step 7 RNO, of ECA-1.2, if the leak is NOT isolated you are to go to ECA-1.1 and per the background document: The operator transfers to E-1, LOSS OF REACTOR OR SECONDARY COOLANT, if the break has been isolated, for further recovery actions. If the break has not been isolated, the operator is sent to ECA-1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, for further recovery actions since there will be no inventory in the sump.

Examination:	RO 2008 Ginna NRC	**Question #:**	80	Rev. 2	Level: SRO
Lesson Plan:	REC11C	ECA-1.1, Loss Of Emergency Coolant Recirculation			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in ECA-1.1, Loss of Emergency Coolant Recirculation			
Category:	Tier 1 / Group 1	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	W/E11 Loss of Emergency Coolant Recirc. EA2.1 - Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation): Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	**ROI:** 3.4	**SROI:** 4.2		
Technical References:	ECA-1.2, LOCA OUTSIDE CONTAINMENT pg. 9 ECA-1.2 Background document pg. 15	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	C000.1125		
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 81

Plant conditions are as follows:

- The unit tripped from rated power and is currently in E-0, Reactor Trip or Safety Injection.
- RCS pressure – 1780 psig and lowering.
- Pressurizer Level – 2 % and lowering.
- Containment pressure – 0.1 psig and stable.
- "A" S/G Pressure – 800 psig and lowering slowly.
- "B" S/G Pressure – 450 psig and lowering quickly.
- "A" S/G Water Level – 8% (NR) and slowly rising.
- "B" S/G Water Level – 300 inches (WR) and lowering.
- R-9, Letdown Rad Monitor is in alarm.
- R-32, "B" Main Steam Line Rad Monitor is slowly rising.
- R-2, Containment Rad Monitor is rising slowly but not in alarm.
- R-7, Incore Detector Area Rad Monitor is rising slowly but not in alarm.

Which of the following procedures will the CRS transition to, **FIRST** to mitigate the consequences of the given plant conditions?

a. E-3, Steam Generator Tube Rupture

b. E-2, Faulted Steam Generator Isolation

c. ECA-1.2, LOCA Outside Containment

d. E-1, Loss Of Reactor or Secondary Coolant

Answer:

b. E-2, Faulted Steam Generator Isolation

Explanation:

Incorrect	a.	SI will be actuated on low S/G press at 514 psig and should be terminated but only after first going to E-2 to isolate the steam leak.
Correct	b.	With S/G level and pressure trending down E-2 is the correct procedure to mitigate the given plant conditions.
Incorrect	c.	Leak is outside containment but it is a Steam leak not a coolant leak, ECA-1.2 will provide no assistance. With the R-9 alarm may think there is a leak outside Cnmt.
Incorrect	d.	Has all the classic signs of a LOCA and with Rad alarms in various areas it maybe mistaken for a coolant leak.

Examination:	RO 2008 Ginna NRC	**Question #:**	81	Rev. 4	Level: SRO
Lesson Plan:	REP02C	E-2, Faulted Steam Generator Isolation			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in E-2, Faulted Steam Generator Isolation.			
Category:	Tier 1 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	W/E12 - Steam Line Rupture - Excessive Heat Transfer 2.1.7 - Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.	**ROI:**	4.4	**SROI:**	4.7
Technical References:	LP REP02C, E-2, Faulted Steam Generator Isolation pgs. 8,9 E-2, Faulted Steam Generator Isolation pg. 2	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 82

Plant conditions occurred as follows:

- The plant is in a Refueling outage.
- Core Alterations were in progress.
- Fuel movements were taking place in Containment and the Auxiliary Building.
- The Refueling SRO reported one of the spent fuel assemblies dropped on to the core and damaged several fuel assemblies.
- R-2, Containment Area Monitor is in alarm and reading off scale high.
- R-29/30, Containment Radiation Monitors are reading 125 R/hr and slowly trending up.

Which one of the following emergency classifications would apply for the current plant conditions?

a. Unusual Event

b. Alert

c. Site Area Emergency

d. General Emergency

Answer:

b. Alert

Explanation:

Incorrect	a.	Meets the requirements of 5.3.1 but conditions are given for an Alert.
Correct	b.	Required per EAL 2.4.2.
Incorrect	c.	Meets the requirements of 2.3.2 but only applicable in Modes 1-4 not in Mode 6.
Incorrect	d.	May select GE due to dropped fuel and rad levels. Incorrect EAL call at this time. Meets rad requirements for 4.1.6, but 4.1.6 is only applies in Modes 1-4.

Examination:	RO 2008 Ginna NRC	**Question #:**	82	Rev. 1	Level: SRO
Lesson Plan:	RSC02C	Classification, Implementation and Notification			
Objective(s):	3.00	Using the appropriate EPIP procedure and a given set of plant conditions: Classify the event.			
Category:	Tier 1 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000036 Fuel Handling Accident 2.4.41 - Knowledge of the emergency action level thresholds and classifications.	**ROI:**	2.9	**SROI:**	4.6
Technical References:	EPIP-1.0, Ginna Station Event Evaluation and Classification Pgs. 7,8,14,19	**References provided during Exam:**	EPIP-1.0, Ginna Station Event Evaluation and Classification		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 83

Plant conditions are as follows:

- The reactor is currently at 100% power.
- The unit has been operating with a Steam Generator Tube Leak (SGTL) of 60 Gallon per Day (GPD) in the "B" S/G for the last three months.
- Currently doing four hour leak determinations.
- PPCS alarm R-47G $\geq$ 75 GPD just alarmed.
- A review of leak trend shows the following:
  - 1300 - leak started increasing above 60 GPD
  - 1301 - 175 GPD
  - 1302 - 65 GPD
  - 1310 - 135 GPD
  - 1311 - 75 GPD
  - 1320 - 85 GPD
  - 1330 - 90 GPD
  - 1345 - 95 GPD and steady (current time)
  - All radiation monitors operable and consistently trend with R-47G

What procedure will be used and what actions are required based on the current plant conditions?

a. Remain in AP-SG.1, Steam Generator Tube Leak and increase monitoring the "B" S/G to one hour intervals for leakage.

b. Remain in AP-SG.1, Steam Generator Tube Leak and check S/G leak rate remains greater than 75 GPD for greater than one hour.

c. Commence a Load Reduction per AP-TURB.5, Rapid Load Reduction because power must be reduced to less than 50 % power in the next hour.

d. Commence a plant shutdown per O-2.1, Normal Shutdown to Hot Shutdown because the unit is required to be in Mode 3 within the next six hours.

Answer:

c. Commence a Load Reduction per AP-TURB.5, Rapid Load Reduction because power must be reduced to less than 50 % power in the next hour.

Explanation:

Incorrect	a.	If misapplies step 4c and goes to RNO column and then will be sent to step 5 and at step 5b will have the operators increase monitoring of the "B" S/G to one hour intervals for leakage.
Incorrect	b.	If misapplies step 4d and goes to RNO column will come to requirement for monitoring S/G tube leakage for >75 GPD for greater than one hour prior to moving / proceeding to step 6.
Correct	c.	This is correct if during evaluation/execution of step 7 that the rise in S/G tube leakage is less than 30 GPD / hr and answers no and goes to RNO to lower power to 50% w/I (1) hour.
Incorrect	d.	This is correct only if during evaluation/execution of step 7 that the rise in S/G tube leakage is less than 30 GPD / hr and answers yes to this step will proceed to "c" then to "d" where: would answer no to "has remained <150 GPD since leak initiation (due to spike at 1301 – 175 GPD).

Examination:	RO 2008 Ginna NRC	**Question #:**	83	Rev. 2	Level: SRO
Lesson Plan:	RAP32C	Steam Generator Tube Leak, AP-SG.1			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement.(AP-SG.1)			
Category:	Tier 1 / Group 2	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	000037 Steam Generator Tube Leak 2.1.20 - Ability to interpret and execute procedure steps.	**ROI:**	4.6	**SROI:**	4.6
Technical References:	AP-SG.1, Steam Generator Tube Leak pgs. 3-10	**References provided during Exam:**	AP-SG.1, Steam Generator Tube Leak		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	B000.1003		
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 84

Plant conditions are as follows:

- The unit is at rated power.
- A liquid radwaste release is in progress.
- R-18, Waste Liquid Radiation Monitor Warning Alarm comes in quickly followed by the High Alarm.

Which one of the following describes the action required?

In accordance with the annunciator procedure verify . . .

a. RCV-018 is closed, contact an AO to resample the tank; and refer to EPIP 1- 16, Radioactive Liquid Release to Lake Ontario or Deer Creek.

b. the tank is recirculating, reverify the lineup is correct, contact an AO to resample the tank; and refer to EPIP 1-1, Unusual Event.

c. the tank is recirculating, perform a flush of R-18, and refer to EPIP1-0, Ginna Station Event Evaluation and Classification.

d. RCV-018 is closed, submit an A-52.12 for R-18, refer to the ODCM for operability requirements and contact an AO to resample the tank.

Answer:

a. RCV-018 is closed, contact an AO to resample the tank; and refer to EPIP 1- 16, Radioactive Liquid Release to Lake Ontario or Deer Creek.

Explanation:

Correct	a.	All required per AR-RMS-18, R-18 Waste Liquid.
Incorrect	b.	Suppose to verify RCV-018 is closed, tank won't be in recirc as the high alarm also trips the pump. Not required to verify the lineup except if performing a discharge with R-18 inop. Refer to EPIP 1-0 or 1-16 not 1-1.
Incorrect	c.	Suppose to verify RCV-018 is closed, tank won't be in recirc as the high alarm also trips the pump. Referring to EPIP1-0, Ginna Station Event Evaluation and Classification is not required by the ARP.
Incorrect	d.	A-52.12 submittal not required due to R-18 operating correctly and should sample first to prove R-18 is inop. No ODCM entry required.

Examination:	RO 2008 Ginna NRC	**Question #:**	84	Rev. 3	Level: SRO
Lesson Plan:	R3801C	Waste Disposal System			
Objective(s):	R4.03	Given a set of plant conditions and an alarming annunciator associated with the waste disposal system, evaluate plant conditions and determine the appropriate operator response.			
Category:	Tier 1 / Group 2	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	000059 Accidental Liquid RadWaste Rel AA2.05 - Ability to determine and interpret the following as they apply to the Accidental Liquid Radwaste Release: The occurrence of automatic safety actions as a result of a high PRM system signal.	**ROI:**	3.6	**SROI:**	3.9
Technical References:	AR-RMS-18, R-18 Waste Liquid pg. 1 LP R3901C, Radiation Monitoring System pg. 20	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 85

Plant conditions occurred as follows:

- The unit was at 80% power when a Loss of Coolant Accident (LOCA) occurred.
- The Control Room Team has performed the applicable EOPs and are currently performing E-1, Loss of Reactor or Secondary Coolant.
- FR-Z.1, Response To High Containment Pressure is now entered due to a valid Red Path for Containment pressure.

What procedure will govern the use of Containment Sprays and why?

a. FR-Z.1, Response To High Containment Pressure, allows maximum available heat removal system capability in order to reduce containment pressure.

b. E-1, Loss of Reactor or Secondary Coolant, because of a less restrictive criteria, which permits reduced spray pump operation depending on containment pressure and number of emergency fan coolers operating.

c. FR-Z.1, Response To High Containment Pressure, because of a less restrictive criteria, which permits reduced spray pump operation depending on containment pressure and number of emergency fan coolers operating.

d. E-1, Loss of Reactor or Secondary Coolant, allows maximum available heat removal system capability in order to reduce containment pressure.

Answer:

b. E-1, Loss of Reactor or Secondary Coolant, because of a less restrictive criteria, which permits reduced spray pump operation depending on containment pressure and number of emergency fan coolers operating.

Explanation:

Incorrect	a.	This is correct for the procedure and the reason if entry to FR-Z.1 was NOT from E-1 or ECA-1.1. Since entry was from E-1, E-1 is the guiding document for the use of Cnmt Sprays.
Correct	b.	In the procedure for FR-Z.1, Response To High Containment Pressure, it instructs the operators to use the guidance in E-1 if it was in effect. The background document warns the operator that the operation of the containment spray pumps indicated in E-1 takes precedence over that noted in the next step of this procedure. This procedure specifies maximum available heat removal system operability in order to reduce containment pressure. E-1 uses a less restrictive criteria, which permits reduced spray pump operation depending on containment pressure and number of emergency fan coolers operating. The less restrictive criteria for containment spray operation is used in E-1 to conserve RWST water, if possible, by stopping containment spray pumps.
Incorrect	c.	FR-Z.1 allows maximum flow of Cnmt Sprays but since entry was from E-1 FR-Z.1 is not the controlling document.
Incorrect	d.	Correct procedure per guidance given in FR-Z.1, but the incorrect reason. E-1 tries to conserve the RWST for as long as possible and has a less restrictive criteria, which permits reduced spray pump operation depending on containment pressure and number of emergency fan coolers operating.

Examination:	RO 2008 Ginna NRC	**Question #:**	85	Rev. 2	Level: SRO
Lesson Plan:	RFRZ1C	FR-Z.1, Response To High Containment Pressure			
Objective(s):	1.02	State the basis for Cautions, Notes, and Major Action Categories in FR-Z.1, Response To High Containment Pressure.			
Category:	Tier 1 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	W/E14 Loss of CTMT Integrity EA2.2 - Ability to determine and interpret the following as they apply to the (High Containment Pressure): Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	**ROI:** 3.3	**SROI:** 3.8		
Technical References:	FR-Z.1, Response To High Containment Pressure pg. 4 and Background Document pgs. 9,10 E-1, Loss of Reactor or Secondary Coolant	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 86

Plant conditions are as follows:

- The unit has tripped from rated power due to a Loss of Coolant Accident (LOCA).
- RCS pressure – 1450 psig and stable.
- RVLIS Water Level – 65% and stable.
- Containment pressure – 8.5 psig and slowly rising.
- CETs – 810°F and slowly rising.
- RWST level – 70% and lowering.
- Containment sump 78 inches.
- No RCPs are running.
- Containment radiation monitors are in alarm.
- All ECCS systems are running as expected.
- E-0, Reactor Trip or Safety Injection is being performed.

1. What would be the impact on the SI system if a rupture were to occur on the inlet weld to MOV-878A, SI to RCS injection valve?

and

2. Which ONE of the following lists the correct actions and procedure transitions due to the LOCA and the SI rupture?

a. 1. SI Line to RCS Loop "A" – SI flow on FI-925 and pressure on PI-923 would remain stable.
   SI Line to RCS Loop "B" – SI flow on FI-924 would rise and SI pressure on PI-922 would lower.
   2. E-1 to FR-C.2 to E-1 to ES-1.2 to ES-1.3

b. 1. SI Line to RCS Loop "A" – SI flow on FI-925 would rise and SI pressure on PI-923 would lower.
   SI Line to RCS Loop "B" – SI flow on FI-924 and pressure on PI-922 would remain stable.
   2. FR-C.2 to E-0 to E-1 to ECA-1.1

c. 1. SI Line to RCS Loop "A" – SI flow on FI-925 and pressure on PI-923 would remain stable.
   SI Line to RCS Loop "B" – SI flow on FI-924 would rise and SI pressure PI-922 would lower.
   2. FR-C.2 to E-0 to E-1 to ECA-1.1

d. 1. SI Line to RCS Loop "A" – SI flow on FI-925 would rise and SI pressure on PI-923 would lower.
   SI Line to RCS Loop "B" – SI flow on FI-924 and pressure PI-922 would remain stable.
   2. E-1 to FR-C.2 to E-1 to ES-1.2 to ES-1.3

Answer:

a. 1. SI Line to RCS Loop "A" – SI flow on FI-925 and pressure on PI-923 would remain stable.
SI Line to RCS Loop "B" – SI flow on FI-924 would rise and SI pressure on PI-922 would lower.
2. E-1 to FR-C.2 to E-1 to ES-1.2 to ES-1.3

Explanation:

Correct	a.	1. MOV-878A is a normally closed valve that gets SI flow from the "A" SI pump and delivers it to the "B" Hot leg. With the leak on the upstream side where MOV-878B also taps in will result in high flows and lowering pressure on the SI line to the "B" RCS loop. 2. Transition out of E-0 to E-1 prior to start of CSFST monitoring. Once in E-1 CSFST monitoring starts, then go to FR-C.2 then back to E-1, ES-1.2 and ES-1.3. ECA-1.1 will be entered if the leak can't be stopped. Any transition to ECA-1.1 will only happen after ES-1.3 is entered.
Incorrect	b.	1. This would be correct if MOV-878A was on a common line that went to the "A" RCS loop but it does not, it is on a common line that goes to the "B" RCS loop. 2. Not monitoring CSFST in E-0 yet, so transition would only happen when in E-1. Would come back to E-0 when FR-C.2 was completed then go to E-1. Would make sense to go to ECA-1.1 from E-1 but none of the requirements are met to go to ECA-1.1, have SW, CCW, RHR loops all available and there is a leak in the AB but not a coolant leak.
Incorrect	c.	1. MOV-878A is a normally closed valve that gets SI flow from the "A" SI pump and delivers it to the "B" Hot leg. With the leak on the upstream side where MOV-878B also taps in will result in high flows and lowering pressure on the SI line to the "B" RCS loop. 2. Not monitoring CSFST in E-0 yet, so transition would only happen when in E-1. Would come back to E-0 when FR-C.2 was completed then go to E-1. Would make sense to go to ECA-1.1 from E-1 but none of the requirements are met to go to ECA-1.1, have SW, CCW, RHR loops all available and there is a leak in the AB but not a coolant leak.
Incorrect	d.	1. This would be correct if MOV-878A was on a common line that went to the "A" RCS loop but it does not, it is on a common line that goes to the "B" RCS loop. 2. Transition out of E-0 to E-1 prior to start of CSFST monitoring. Once in E-1 CSFST monitoring starts, then go to FR-C.2 then back to E-1, ES-1.2 and ES-1.3. ECA-1.1 will be entered if the leak can't be stopped. Any transition to ECA-1.1 will only happen after ES-1.3 is entered.

Examination:	RO 2008 Ginna NRC	**Question #:**	86	Rev. 4	Level: SRO
Lesson Plan:	R2701C	Emergency Core Cooling System (ECCS)			
Objective(s):	1.07	Given system conditions, describe the design features of the Emergency Core Cooling System, to include set points, interlocks, and the related automatic action(s) for the following components, instrumentation and/or processes. b. Safety Injection Pumps.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	006 Emergency Core Cooling A2.11 - Ability to (a) predict the impacts of the following malfunctions or operations on the ECCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture of ECCS header	**ROI:**	4.0	**SROI:**	4.4
Technical References:	LP R2701C, ECCS pg. 30 F-0.2, Att-14.5 E-0, E-1, ES-1.2, ES-1.3, FR-C.2, ECA-1.1, AR-B-8	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 87

Plant conditions are as follows:

- The unit has been operating at 100% power for two weeks following a refueling Outage.
- Engineering has called to inform you of an error that was made when the Pressurizer Safety Valve setpoints were adjusted. The error was just discovered and Engineering has determined the Pressurizer Safety valves will now lift at: PCV-434 - 2540 psig
PCV-435 - 2580 psig

1. What is the correct Technical Specification action?

and

2. What is the bases for the Pressurizer Safety Valve lift setpoint?

a. 1. Restore valve to Operable status within 15 minutes.
2. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the RCP locked rotor accident.

b. 1. Restore valve to Operable status within 15 minutes.
2. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the RCP sheared shaft accident.

c. 1. Restore valve to Operable status within 30 minutes.
2. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the RCP sheared shaft accident.

d. 1. Restore valve to Operable status within 30 minutes.
2. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the RCP locked rotor accident.

Answer:

a. 1. Restore valve to Operable statue within 15 minutes.
   2. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure for all anticipated transients except for the RCP locked rotor accident.

Explanation:

Correct	a.	1. Right time per LCO 3.4.10 action A.1 2. Per the bases for LCO 3.4.10: The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure for all transients except locked rotor accidents which has an allowed limit of 120% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.
Incorrect	b.	1. Right time per LCO 3.4.10 action A.1 2. Sheared shaft is not the exception given in the bases.
Incorrect	c.	1. Only 15 minutes is allowed per LCO 3.4.10 action A.1 2. Sheared shaft is not the exception given in the bases.
Incorrect	d.	1. Only 15 minutes is allowed per LCO 3.4.10 action A.1 2. Per the bases for LCO 3.4.10: The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure for all transients except locked rotor accidents which has an allowed limit of 120% of design pressure.

Examination:	RO 2008 Ginna NRC	**Question #:**	87	Rev. 2	Level: SRO
Lesson Plan:	R1901C	Pressurizer Pressure and Level Control			
Objective(s):	1.12	Given a set of plant conditions for the Pressurizer Pressure and Level Control System, or the Reactor Vessel Overpressure Protection System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): a. Identify action statements of less than one hour. b. Describe the basis of any (LCO) and/or Safety Limit.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	010 Pressurizer Pressure Control 2.2.40 - Ability to apply technical specifications for a system.	**ROI:**	3.4	**SROI:**	4.7
Technical References:	ITS 3.4.10 and bases. pgs. B 3.4.10-2,3	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(2)			

Question 88

Plant conditions are as follows:

- The unit is at 90% power.
- Channel "2" of OP-delta T is inoperable and has been in the trip condition for the last 12 hours.
- It has been determined that Channel "3" OP-delta T surveillance must be performed.
- The note in ITS 3.1.1 required action D.1 that says, "The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels", is used to support the I&C surveillance testing on Channel "3" OP-delta T.

Two hours into the surveillance I/C reports the as-found set point for Channel "3" OP-delta T was not within the required band and cannot be adjusted into the proper band.

1. When must Channel "3" OP-delta T be declared inoperable?

and

2. What is the bases for the OP-delta T trip function?

a. 1. I&C has (2) hours to fix and retest Channel "3" OP-delta T before the train must be declared inoperable.
   2. Ensures that the design limit departure from nucleate boiling ratio (DNBR) is met.

b. 1. Immediately declare Channel "3" OP-delta T inoperable.
   2. Ensures that the design limit departure from nucleate boiling ratio (DNBR) is met.

c. 1. I&C has (2) hours to fix and retest Channel "3" OP-delta T before the train must be declared inoperable.
   2. Ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding failure).

d. 1. Immediately declare Channel "3" OP-delta T inoperable.
   2. Ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding failure).

Answer:

d. 1. Immediately declare Channel "3" OP-delta T inoperable.
2. Ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding failure).

Explanation:

Incorrect	a.	1. The surveillance testing is done and must be declared inop immediately. Corrective maintenance and trouble shooting is now required to fix the channel. The note only allows testing to verify the channel is working correctly. 2. This is the ITS bases for OT-delta T trip set point.
Incorrect	b.	1. The channel no longer meets the surveillance requirements and is therefore inop 2. This is the ITS bases for OT-delta T trip set point.
Incorrect	c.	1. The surveillance testing is done and must be declared inop immediately. Corrective maintenance and trouble shooting is now required to fix the channel. The note only allows testing to verify the channel is working correctly. 2. This is the ITS bases for OP-delta T trip set point.
Correct	d.	1. The channel no longer meets the surveillance requirements and is therefore inop. 2. This is the ITS bases for OP-delta T trip set point.

Examination:	RO 2008 Ginna NRC	**Question #:**	88	Rev.3	Level: SRO
Lesson Plan:	R3501C	Reactor Protection System			
Objective(s):	1.12	Given a set of plant conditions for the Reactor Protection System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): b. Describe the basis of any (LCO) and/or Safety Limit.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	012 Reactor Protection 2.2.40 - Ability to apply technical specifications for a system	**ROI:** 3.4	**SROI:** 4.7		
Technical References:	ITS 3.3.1 pgs. 3.3.1-2,10 ITS 3.3.1 Bases pgs. B3.3.1-13,14,31	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(2)			

Question 89

Plant conditions are as follows:

- The unit has experienced a large break Loss of Coolant Accident (LOCA).
- The Control Room team has properly implemented the EOPs as required.
- The plant is in Cold Leg Recirculation with the following conditions:
  - "A" RHR pump is running and "B" RHR pump is available.
  - All of the Containment Recirc Fans have failed to start and won't run.
  - "B" CCW pump is running and "A" CCW pump is OOS.
  - "C" SI pump is running with "A" and "B" SI pumps available.
  - "A" CS pump is running and "B" CS pump is available.
  - Total RHR flow is 1500 gpm and stable.
  - Containment Pressure is 48 psig and stable.
  - RHR Suction Temperature is 285°F.

The "B" CCW pump has just tripped due to an issue with the breaker and cannot be restarted.

How will the plant respond and what actions will need to be taken based on plant response?

a. "A" RHR pump will begin to show signs of cavitation, per ES-1.3, Cold Leg Recirculation start another CS pump to provide more flow through the RHR pump and to lower containment pressure.

b. RHR suction temperature will begin to rise, per ECA-1.1, Loss of Emergency Coolant Recirculation, start the "B" RHR pump and raise RHR flow.

c. "A" RHR pump will begin to show signs of cavitation, per ECA-1.1, Loss of Emergency Coolant Recirculation, lower RHR flow to correct the RHR pump cavitation issue.

d. RHR suction temperature will begin to rise, secure the running CS pump per ES-1.3, Cold Leg Recirculation due to limitations on the CS pump.

Answer:

d. RHR suction temperature will begin to rise, secure the running CS pump per ES-1.3, Cold Leg Recirculation due to limitations on the CS pump.

Explanation:

Incorrect	a.	With 48 psig in Cnmt and RHR lined up to the B sump there will be NO indications of cavitation but it is plausible with loss of CCW for the student to think RHR will cavitate. ES-1.3 calls for the running CS pump to be secured due to RHR suction temp rising.
Incorrect	b.	RHR suction temperature will begin to rise due to loss of CCW, with nothing else providing cooling (CRFs not running). Entry into ECA-1.1would be correct however SI, CS and RHR pumps would be secured not started
Incorrect	c.	With 48 psig in Cnmt and RHR lined up to the B sump there will be NO indications of cavitation but it is plausible with loss of CCW for the student to think RHR will cavitate. The action to correct the cavitation is similar to the actions taken when the RCS loops are drained and RHR cavitation occurs.
Correct	d.	RHR suction temperature will begin to rise due to loss of CCW, with nothing else providing cooling (CRFs not running). ES-1.3 calls for the running CS pump to be secured due to RHR suction temp rising.

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<tr><td>K/A:</td><td colspan="4">026 Containment Spray
A2.07 - Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of containment spray pump suction when in recirculation mode, possibly caused by clogged sump screen, pump inlet high temperature exceeded cavitation, voiding), or sump level below cutoff (interlock) limit</td><td>ROI:
SROI:</td><td>3.6
3.9</td></tr>
<tr><td>Technical References:</td><td colspan="2">ES-1.3, Transfer To Cold Leg Recirculation pg. 13
Att 28.0, Att Cnmt Spray pg. 1
Fig-21 Fig Cnmt Spray Restart Criteria pg. 1
ECA-1.3, B Sump Blockage</td><td>References provided during Exam:</td><td colspan="3">Fig-21 Fig Cnmt Spray Restart Criteria pg. 1
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<tr><td>10CFR 55 Content:</td><td colspan="2">55.41</td><td colspan="4">55.43(b)(5)</td></tr>
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Question 90

Plant conditions are as follows:

- The plant is in Mode 3.
- Tavg is 547°F and stable.
- RCS pressure is 2235 psig and stable.
- STP-O-12.2, Emergency Diesel Generator "B", monthly surveillance test is underway.
- "B" D/G has been loaded per the surveillance procedure for about 30 minutes.

At this time, all of Off-site power is lost coincident with a large break Loss of Coolant Accident.

What is the expected plant response if all equipment operates as designed and what actions must be directed in response to the loss of Off-site power coincident with the LOCA?

a. Initially, the voltage and frequency of Emergency Diesel Generator "B" will be high
Per STP-O-12.2, immediately place the UNIT/PARALLEL switch to UNIT.
Then adjust voltage as required to achieve 480 volts and adjust frequency to 60.0 HZ.

b. Initially, the voltage and frequency of Emergency Diesel Generator "B" will be high.
Per STP-O-12.2, adjust voltage as required to achieve 480 volts.
When load sequencing is completed, place the UNIT/PARALLEL switch to UNIT and adjust frequency to 60.0 HZ.

c. Initially, the Emergency Diesel Generator "B" output breakers to Buses 16 and 17 will open.
Per AP-ELEC.14/16 and AP-ELEC.17/18, place the UNIT/PARALLEL switch to UNIT first, and then close the D/G supply breakers to Buses 16 and 17.
Then adjust voltage as required to achieve 480 volts and adjust frequency to 60.0 HZ.

d. Initially, the Emergency Diesel Generator "B" output breakers to Buses 16 and 17 will open.
Per AP-ELEC.14/16 and AP-ELEC.17/18, close the D/G supply breakers to Buses 16 and 17 first, then when load sequencing is completed, place the UNIT/PARALLEL switch to UNIT.
Then adjust voltage as required to achieve 480 volts and adjust frequency to 60.0 HZ.

Answer:

b. Initially, the voltage and frequency of Emergency Diesel Generator "B" will be high.
Per STP-O-12.2, adjust voltage as required to achieve 480 volts.
When load sequencing is completed, place the UNIT/PARALLEL switch to UNIT and adjust frequency to 60.0 HZ.

Explanation:

Incorrect	a.	Per Att. 2 of STP-O-12.2, Emergency Diesel Generator "B" initially, the voltage and frequency of Emergency Diesel Generator B will be high and only after the loads have sequenced should the UNIT/PARALLEL switch be placed in UNIT.
Correct	b.	As per Att. 2 of STP-O-12.2, Emergency Diesel Generator "B".
Incorrect	c.	Breakers will not open, entry into AP ELECT.14/16 or 17/18 is not required.
Incorrect	d.	Breakers will not open, entry into AP ELECT.14/16 or 17/18 is not required.

Examination:	RO 2008 Ginna NRC	**Question #:**	90	Rev. 2	Level: SRO
Lesson Plan:	R0801C	Diesel Generator System			
Objective(s):	1.09	Given a specific fundamental principle, performance test, limits and/or precautions for the Diesel Generator System or component, describe the reason/basis and the effect on the operations of the system or component.			
Category:	Tier 2 / Group 1	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	4		
K/A:	064 Emergency Diesel Generator A2.16 - Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of offsite power during full-load testing of ED/G	**ROI:**	3.3	**SROI:**	3.7
Technical References:	STP-O-12.2, Emergency Diesel Generator "B" pg, 44	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 91

Plant conditions are as follows:

- The unit is at 90% power.
- All control systems in there normal alignment, and no equipment is out of service.
- The following annunciators come in:
    - o F-28, Pressurizer High Level Channel Alert 87%
    - o F-4, Pressurizer Level Deviation -5 NORMAL +5
    - o A-4, Regen Hx Outlet Hi Temp 395°F

1. What was the common cause for these annunciators?

and

2. What procedure will provide the proper guidance to mitigate the current plant conditions?

a. 1. There is a steam space leak in the Pressurizer, which has caused Pressurizer level to swell.
   2. Enter AP-RCS.1, Reactor Coolant Leak.

b. 1. LT-426 Pressurizer Level Failed High causing the charging pump in auto to go to minimum speed.
   2. Enter ER-INST.1, Reactor Protection Bistable Defeat after Instrumentation Loop Failure.

c. 1. There is a leak on the charging header down stream of HCV-142, Charging Backpressure Control Valve, causing HCV-142 to throttle closed.
   2. Enter AP-RCS.1, Reactor Coolant Leak.

d. 1. LT-428 Pressurizer Level Failed High causing the charging pump in auto to go to minimum speed.
   2. Enter ER-INST.1, Reactor Protection Bistable Defeat after Instrumentation Loop Failure.

Answer:

d. 1. LT-428 Pressurizer Level Failed High causing the charging pump in auto to go to minimum speed.
2. Enter ER-INST.1, Reactor Protection Bistable Defeat after Instrumentation Loop Failure.

Explanation:

Incorrect	a.	Steam space leaks will result in Pzr level rising but only after level has dropped and a steam void is drawn in the Rx head. No RCS leak therefore no entry into AP-RCS.1 is needed but if it were a leak it would be the correct AP to enter.
Incorrect	b.	The controlling channel, LT-428 is in service per the stem, LT-426 is not the controlling channel at this time. Entry into ER-INST.1, Reactor Protection Bistable Defeat after Instrumentation Loop Failure is the correct procedure to enter.
Incorrect	c.	Explains why A-4, Regen Hx Outlet Hi Temp 395°F is in but not the other alarms. No other indications of a leak so entry into AP-RCS.1 is not required.
Correct	d.	The controlling channel, LT-428 has failed high which caused the F-4 and F-28 alarms. A-4 came in because the auto charging pump seeing a high Pzr level slowed to minimum and reduced the cooling flow to the RHX. Entry into ER-INST.1 will provide guidance to cope with the instrument failure.

Examination:	RO 2008 Ginna NRC	**Question #:**	91	Rev. 3	Level: SRO
Lesson Plan:	RIC03C	Pressurizer Level Channels			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Pressurizer Level Channel Instrumentation will have on related plant system(s) and/or plant operations. b. Plant operation.			
Category:	Tier 2 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	011 Pressurizer Level Control A2.10 - Ability to (a) predict the impacts of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Failure of PZR level instrument - high	**ROI:**	3.4	**SROI:**	3.6
Technical References:	LP RIC03C, Pressurizer Level Channels pgs. 7,8.9 ER-INST.1, Reactor Protection Bistable Defeat after Instrumentation Loop Failure pgs. 8,9	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 92

Plant conditions occurred as follows:

- The unit is at 80% power.
- All control systems in their normal alignment.
- PI-429, Pressurizer Pressure transmitter failed last shift and its required Technical Specification actions have been completed while awaiting repairs.
- PI-449, Pressurizer Pressure instrument output signal is interrupted due to a ground and fails low.

What is the plant response and what procedure is entered in response to plant conditions?

a. Low Pressurizer pressure Safety Injection, Enter E-0, Reactor Trip or Safety Injection.

b. OP Delta T runback with Rod Stop channel alert, enter AP-TURB.2, Turbine Load Reduction.

c. Low Pressurizer pressure Reactor trip, Enter E-0, Reactor Trip or Safety Injection.

d. OT Delta T runback with Rod Stop channel alert, enter AP-TURB.2, Turbine Load Reduction.

Answer:

c. Low Pressurizer pressure Reactor trip, Enter E-0, Reactor Trip or Safety Injection..

Explanation:

Incorrect	a.	No SI signal, SI signals (2/3 LOGIC) come from PI-429,430,431. Correct procedure to enter based on SI.
Incorrect	b.	OP and OT delta–Ts frequently confused. Pressure has no input so OP Delta T will be unaffected.
Correct	c.	2/4 logic met Reactor will trip and entry into E-0 required.
Incorrect	d.	Will get signals and alarms for an OT Delta T runback with rod stop channel alert however there will also be a Reactor Trip which requires entry into E-0.

Examination:	RO 2008 Ginna NRC	**Question #:**	92	Rev. 3	Level: SRO
Lesson Plan:	RIC02C	Pressurizer Pressure Channels			
Objective(s):	1.06	Predict the effect that a loss or malfunction in the Pressurizer Pressure Channels will have on related plant system(s) and/or plant operations. a. Reactor Protection System			
Category:	Tier 2 / Group 2	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	016 Non-nuclear Instrumentation A2.03 - Ability to (a) predict the impacts of the following malfunctions or operations on the NNIS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Interruption of transmitted signal.	**ROI:**	3.0	**SROI:**	3.3*
Technical References:	P-10, pg. 13	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 93

Liquid Radwaste discharges are governed by Technical Specifications section 5.5.4, Radioactive Effluent Controls Program which conforms to _______(1)_______ for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents _______(2)_______ .

The program shall be contained in the_______(3)_______ and shall be implemented by procedures.

a. 1. 10CFR 50
   2. to < 25 mRem TEDE per year
   3. TRM

b. 1. 10CFR 50
   2. as low as reasonably achievable
   3. ODCM

c. 1. 10CFR 100
   2. to < 25 mRem TEDE per year
   3. TRM

d. 1. 10CFR 100
   2. as low as reasonably achievable
   3. ODCM

Answer:

b. 1. 10CFR 50
   2. as low as reasonably achievable
   3. ODCM

Explanation:

Incorrect	a.	Exposure is ALARA, < 25 mRem TEDE per year is the limit for exposure due to all site releases and from uranium cycle sources, per 1.3 of the ODCM. The program is contained within the ODCM not the TRM.
Correct	b.	As per ITS section 5.5.4: This program conforms to 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program shall be contained in the ODCM, shall be implemented by procedures, and shall include remedial actions to be taken whenever the program limits are exceeded.
Incorrect	c.	10 CFR 100 are the accident limits which is incorrect but plausible. Exposure is ALARA, < 25 mRem TEDE per year is the limit for exposure due to all site releases and from uranium cycle sources, per 1.3 of the ODCM. The program is contained within the ODCM not the TRM.
Incorrect	d.	10 CFR 100 are the accident limits which is incorrect but plausible.

Examination:	RO 2008 Ginna NRC	**Question #:**	93	Rev. 2	Level: SRO
Lesson Plan:	R3901C	Radiation Monitoring System			
Objective(s):	1.12	Given a set of plant conditions for the Radiation Monitoring System, perform the following in accordance with (Technical Specifications, TS Bases, TRM, COLR, PTLR, ODCM): a. Describe the basis of any (LCO) and/or Safety Limit.			
Category:	Tier 2 / Group 2	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	068 Liquid Radwaste 2.2.38 - Knowledge of conditions and limitations in the facility license.	**ROI:**	3.6	**SROI:**	4.5
Technical References:	ITS section 5.5.4, Radioactive Effluent Controls Program pg. 5.5-2 ODCM pg.10	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(1)			

Question 94

The following conditions exist:

- The plant is in mode 4.
- It is Labor Day, dayshift.
- The shift crew composition is at minimum shift manning due to the Holiday.
- The CO becomes seriously ill and must be taken to the hospital.
- There are four hours left until shift change.

As the Shift Manager, what action is required in accordance with Technical Specifications?

a. Immediate action must be taken to obtain a replacement reactor operator within a maximum of one hour.

b. Immediate action must be taken to obtain a replacement reactor operator within a maximum of two hours.

c. Immediate action must be taken to obtain a replacement reactor operator within a maximum of three hours.

d. Shift turnover occurs before action is required. Action should be taken to find a replacement, but is not required.

Answer:

b. Immediate action must be taken to obtain a replacement reactor operator within a maximum of two hours.

Explanation:

Incorrect	a.	ITS requires that immediate action be taken to find a replacement. The position cannot remain unmanned for greater than 2 hours.
Correct	b.	ITS requires that immediate action be taken to find a replacement. The position cannot remain unmanned for greater than 2 hours.
Incorrect	c.	ITS requires that immediate action be taken to find a replacement. The position cannot remain unmanned for greater than 2 hours.
Incorrect	d.	ITS requires that immediate action be taken to find a replacement. The position cannot remain unmanned for greater than 2 hours.

Examination:	RO 2008 Ginna NRC	**Question #:**	94	Rev. 2	Level: SRO
Lesson Plan:	RAD23C	Operations Manual, Operations Section Duties, Shift Scheduling, Turnover			
Objective(s):	3.01	Describe normal shift scheduling requirements including rotation schedule, overtime use, and shift assignments as outlined in Shift Scheduling and Watch Standing Requirements, OPS-SHIFT-SCHEDULE.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	M/F	**Difficulty Level:**	2		
K/A:	2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	**ROI:**	2.9*	**SROI:**	3.9
Technical References:	ITS 5.2.2.a Plant Staff pg. 5.2-2 OPS-SHIFT-ORG 3.1.2, 3.2.1 pgs. 2,3	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	Seabrook Unit 1 2003 NRC Exam		
10CFR 55 Content:	55.41	55.43(b)(1)			

Question 95

The following conditions exist:

- The plant is performing PT-34.1, Initial Criticality and Low-Power Physics Testing with DRWM
- All Technical Specification exceptions for Physics Testing are in effect.
- Plant conditions are as follows:
  - Pressurizer pressure – 2210 psig
  - Pressurizer water level – 20 %
  - Both RCS Loop Tavg – 535°F
  - Thermal Power – 6%

For these given conditions, what is the required Technical Specification action?

a. Immediately lower power to 4% Thermal Power.

b. Restore RCS Tavg to within limits within 30 minutes.

c. Immediately open the Reactor trip breakers.

d. Restore Pressurizer Pressure to within limits within 30 minutes.

Answer:

c. Immediately open the Reactor trip breakers.

Explanation:

Incorrect	a.	ITS 3.1.8 Action B.1 requires the Rx trip breakers to be opened immediately if thermal power is not within the limit (limit is < or = to 5% RTP). With the requirement to lower power less than the limits lends credibility.
Incorrect	b.	Tavg is allowed to be as low as 530°F during Physics Testing so the limit is not violated however the time allowed to restore the parameter is the same as listed in 3.4.2, RCS Minimum Temp for Criticality.
Correct	c.	As per ITS 3.1.8 Action B.1 Rx trip breakers to be opened immediately if thermal power is not within the limit (limit is < or = to 5% RTP).
Incorrect	d.	Pzr pressure is lower than normal but within the required band. The time allowed to restore the parameter is the same as listed in 3.4.3, RCS Press and Temp Limits.

Examination:	RO 2008 Ginna NRC	**Question #:**	95	Rev. 3	Level: SRO
Lesson Plan:	ROP01C	Plant From Hot Shutdown To Steady Load O-1.2			
Objective(s):	1.05	Outline the major evolutions performed during O-1.2			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	C/A	**Difficulty Level:**	4		
K/A:	2.1.37 - Knowledge of procedures, guidelines, or limitations associated with reactivity management.	**ROI:**	4.3	**SROI:**	4.6
Technical References:	ITS 3.1.8, 3.4.2, 3.4.3	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(2)			

Question 96

Plant conditions are as follows:

- The unit is at rated power.
- You are standing watch as the CRS.
- It is about 1900 on a Tuesday.
- You have just been requested to review a Condition Report (CR) that requires corrective maintenance that is for the loss of the "A" Diesel Generator which has placed the unit in a (7) day shutdown LCO.
- The CR was forwarded to you because the Shift Manager (SM) is in a meeting.

What are your actions and when can the maintenance start?

a. You can perform an initial review however the SM must approve.
Then per CNG-MN-4.01-1001, Work Order Execution and Closure Process, once the maintenance work package is completely written and approved the corrective work can start.

b. You can perform an initial review however the SM must approve.
Then per CNG-MN-4.01-1002, Work Order Screening and Prioritization, the corrective work can start immediately and the work package can be put together and approved while the work is being done.

c. You as the CRS can complete the review.
Then per CNG-MN-4.01-1001, Work Order Execution and Closure Process, the corrective work can start immediately and the work package can be put together and approved while the work is being done.

d. You as the CRS can complete the review
Then per CNG-MN-4.01-1002, Work Order Screening and Prioritization, once the maintenance work package is completely written and approved the corrective work can start.

Answer:

d. You as the CRS can complete the review
Then per CNG-MN-4.01-1002, Work Order Screening and Prioritization, once the maintenance work package is completely written and approved the corrective work can start

Explanation:

Incorrect	a.	SM approval is not required. CNG-MN-4.01-1002, Work Order Screening and Prioritization allows any SRO to review the CR if the SM is unavailable. The process for work package/work would be correct if it were for a priority "E" - Emergency Maintenance screening however the work is Pri-2.
Incorrect	b.	SM approval is not required. CNG-MN-4.01-1002, Work Order Screening and Prioritization allows any SRO to review the CR if the SM is unavailable. As a priority-2 work package must be completed before work begins per, CNG-MN-4.01-1002, Work Order Screening and Prioritization.
Incorrect	c.	The CRS can review the CR in the absence of the SM. The process for work package/work would be correct if it were for a priority "E" - Emergency Maintenance screening however the work is Pri-2.
Correct	d.	The CRS can review the CR in the absence of the SM. As a priority-2 work package must be completed before work begins per, CNG-MN-4.01-1002, Work Order Screening and Prioritization.

Examination:	RO 2008 Ginna NRC	**Question #:**	96	Rev. 4	Level: SRO
Lesson Plan:	RAD06C	WR & TR Procedures			
Objective(s):	2.06	State who initially reviews each WR/TR in an effort to determine priority for performing the work.			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.2.19 - Knowledge of maintenance work order requirements.	**ROI:**	2.3	**SROI:**	3.4
Technical References:	CNG-MN-4.01-1001, Work Order Execution and Closure Process CNG-MN-4.01-1002, Work Order Screening and Prioritization pg. 7,8	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 97

Plant conditions occurred as follows:

- The Plant was operating at 100% power with all systems in a normal configuration when a transient occurred that resulted in a Reactor Trip and Safety Injection.
- E-0, Reactor Trip or Safety Injection, is in progress.

The conditions (2) minutes after the trip are as follows:

- RCS Tavg - 500°F and lowering at a constant rate.
- RCS/Pressurizer Pressure - 1850 psig and lowering.
- Both S/G Pressures - 500 psig and lowering.
- Both S/G Water Levels - 6% Narrow Range and slowly lowering.
- Both MSIV's are closed.
- Both MFP's are tripped.
- Both MDAFW pumps are running and delivering ~220 gpm each.
- TDAFW pump is running.
- CTMT Pressure is 0.3 psig and stable.

As the CRS, what procedures will be utilized to mitigate this event and what major action will be directed to mitigate the impact on the reactor?

a. E-2, Faulted Steam Generator Isolation, then ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.
Isolate the TDAFW pump and Reduce AFW Flow to 50 gpm to each Steam Generator to minimize RCS cooldown.

b. E-1, Loss of Reactor or Secondary Coolant, then ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.
Isolate the TDAFW pump and Reduce AFW Flow to 50 gpm to each Steam Generator to minimize RCS cooldown.

c. E-1, Loss of Reactor or Secondary Coolant, then ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.
Control AFW Flow to maintain narrow range level between 7% and 50% in both Steam Generators to ensure maintenance of the secondary heat sink.

d. E-2, Faulted Steam Generator Isolation, then ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.
Control AFW Flow to maintain narrow range level between 7% and 50% in both Steam Generators to ensure maintenance of the secondary heat sink.

Answer:

a. E-2, Faulted Steam Generator Isolation, then ECA-2.1, Uncontrolled Depressurization of Both Steam Generators.
Isolate the TDAFW pump and Reduce AFW Flow to 50 gpm to each Steam Generator to minimize RCS cooldown.

Explanation:

Correct	a.	The condition in the question indicates an uncontrolled depressurization of both S/Gs. The transition to E-2 is made upon recognition that the S/G secondary side is not intact as indicated by S/G pressures lowering (Step 15 of E-0). The transition to ECA-2.1 is made at step 2 of E-2 since neither S/G pressure is stable or rising. ECA-2.1 directs the operator pull stop the TDAFP Steam supply and close the pump flow control valves then reduce Feed Flow to 50 gpm in each S/G due to the cooldown rate being > 100°F/hr.
Incorrect	b.	Procedure routing is incorrect. The conditions for entry into E-1 could conceivably be applied due to secondary coolant is being lost and E-1 does cover the loss of secondary coolant and the resulting cooldown. The action given is correct.
Incorrect	c.	Procedure routing is incorrect. The conditions for entry into E-1 could conceivably be applied due to secondary coolant is being lost and E-1 does cover the loss of secondary coolant and the resulting cooldown. The action given would also result in continued cooldown.
Incorrect	d.	Correct procedure routing however the action is not consistent with the guidance in ECA-2.1 Controlling S/G levels would result in continuing to feed the fault and result in a continued cooldown.

Examination:	RO 2008 Ginna NRC	**Question #:**	97	Rev. 3	Level: SRO
Lesson Plan:	REC21C	ECA-2.1, Uncontrolled Depressurization Of Both Steam Generators			
Objective(s):	2.01	Given a set of plant and equipment conditions evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	2.2.44 - Ability to interpret control room indications to verify the status and operation of system, and understand how operator actions and directives affect plant and system conditions.	**ROI:**	4.2	**SROI:**	4.4
Technical References:	E-0, Reactor Trip or Safety Injection pg. 11 E-2, Faulted Steam Generator Isolation pg. 4 ECA-2.1, Uncontrolled Depressurization of Both Steam Generators. pgs. 3,4	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	C000.1479		
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 98

You as the Shift Manager, are authorizing a Gas Decay Tank Release.

The release must be initiated within ________(1)________ hours of the tank sample time with no additions to the tank since the sample was drawn.

If the release is not initiated within the required time limit, ________(2)________ is required to authorize the release.

a. 1. 12
   2. Chemistry Supervisor permission

b. 1. 12
   2. resampling

c. 1. 24
   2. Chemistry Supervisor permission

d. 1. 24
   2. resampling

Answer:

a. 1. 12
   2. Chemistry Supervisors Permission

Explanation:

Correct	a.	As required per CH-RETS-GDT-REL, Gas Decay Tank Release limitation 7.1: A release is required to be initiated within 12 hours after sample time, unless no additions have been made to tank. (Chemistry) Supervisor's permission is required for release if initiation is not started within 12 hours.
Incorrect	b.	This was correct prior to the recent procedure change. The procedure does not require resampling. The SM or others may choose to do so but is not required.
Incorrect	c.	As required per CH-RETS-GDT-REL, Gas Decay Tank Release limitation 7.1 except the time allowed is not 24 hours, it is 12 hours.
Incorrect	d.	This was correct prior to the recent procedure change except the time allowed is not 24 hours, it is 12 hours. The procedure does not require resampling. The SM or others may choose to do so but is not required.

Examination:	RO 2008 Ginna NRC	**Question #:**	98	Rev. 4	Level: SRO
Lesson Plan:	RRC07C	Radiation Control Administrative Procedures			
Objective(s):	3.00	State the responsibilities of the following positions with respect to the control of radiation exposure: c) Shift Manager			
Category:	Tier 3	**Question Source:**	New		
Cognitive level:	M/F	**Difficulty Level:**	3		
K/A:	2.3.6 - Ability to approve release permits.	**ROI:**	2.0	**SROI:**	3.8
Technical References:	CH-RETS-GDT-REL, Gas Decay Tank Release Pg. 4	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**			
10CFR 55 Content:	55.41	55.43(b)(4)			

Question 99

Plant conditions occurred as follows:

- The unit was operating at 100% power with a mini-purge in progress.
- A reactor trip occurred coincident with a loss of off-site power.
- The Control Room Team have just transitioned to E-1, Loss of Reactor or Secondary Coolant, and the following conditions exist:
  - RCS pressure approximately 25 psig and lowering.
  - Mini purge has failed to isolate.
  - R-29 and R-30 read 125 R/hr.
  - R14 on alarm.
  - R14A7 is reading 2 uci/cc.
  - Meteorological conditions are:
    - 33 ft temp 65 degrees F
    - 250 ft temp 66 degrees F
    - Wind direction 135 degrees
    - Wind speed 4 mph

As the Shift Manager acting as the Emergency Coordinator, you have declared a General Emergency and recommend . . .

a. shelter-in-place of ERPA W-1.

b. evacuation of ERPAs W-1,2,3.

c. evacuation of ERPA W-1.

d. shelter-in-place of ERPAs W-1,2,3.

Answer:

c. evacuation of ERPA W-1.

Explanation:

Incorrect	a.	Correct if using att. 2 instead of att.1.
Incorrect	b.	Correct if wind direction used is 325 degrees (180 degrees out from 135 degrees) and using att. 1.
Correct	c.	Correct PARs per EPIP-2.1 and using att. 1.
Incorrect	d.	Correct if wind direction used is 325 degrees (180 degrees out from 135 degrees) and if using att. 2 instead of att.1.

Examination:	RO 2008 Ginna NRC	**Question #:**	99	Rev. 5	Level: SRO
Lesson Plan:	RSC02C	Classification, Implementation, & Notification			
Objective(s):	1.01	Determine the actions required at each of the four emergency classification levels.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	2.4.40 - Knowledge of SRO responsibilities in emergency plan implementation.	**ROI:**	2.7	**SROI:**	4.5
Technical References:	EPIP-2.1, Protective Action Recommendations pgs. 3,7,8	**References provided during Exam:**	EPIP 2-1		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	B340.0024		
10CFR 55 Content:	55.41	55.43(b)(5)			

Question 100

Plant conditions are as follows:

- The unit has experienced a Reactor Trip and Safety Injection due to a Loss of Coolant Accident (LOCA).
- RCS pressure is 800 psig.
- Containment pressure is 10 psig and slowly rising.
- No RCPs are running.
- (4) CETs are reading 1250°F and the remaining CETs are reading 680°F.
- RVLIS level is 45%.
- All S/G narrow range levels are off-scale low.
- Unable to start the MDAFW pumps and the TDAFW pump.
- All SI pumps have tripped on overload.
- MOV-896B, RWST outlet to SI and CS pumps, was found closed and is now open.
- The Control Room Staff is performing E-1, Loss of Reactor or Secondary Coolant.

Given these plant conditions, which ONE of the following identifies the procedure action to mitigate this event?

a. Transition to FR-H.1, Response to Loss Of Secondary Heat Sink, attempt to start SAFW pumps.

b. Transition to FR-C.1, Response to Inadequate Core Cooling, reset overloads and attempt to start SI pumps.

c. Transition to FR-H.1, Response to Loss Of Secondary Heat Sink, open PORVs and Block Valves.

d. Transition to FR-C.1, Response to Inadequate Core Cooling, open PORVs and Block Valves.

Answer:

a. Transition to FR-H.1, Response to Loss Of Secondary Heat Sink, attempt to start SAFW pumps.

Explanation:

Correct	a.	Heat Sink is currently the highest priority red path safety function. AFW pumps are not available so the SAFW pumps would be started to restore flow to the S/Gs.
Incorrect	b.	It is reasonable to believe that the SI pumps would be restarted with the suction path restored, however core cooling is only in an orange path since a minimum of 5 CETs must be indicating > 1200 degrees for a red path condition per the background document, and the remaining CETs would only support an orange path condition, therefore FR-C.1 is not the correct procedure to implement at this time.
Incorrect	c.	Heat Sink is currently the highest priority red path safety function; however, a start of the SAFW pumps would be attempted prior to resorting to feed and bleed cooling.
Incorrect	d.	Cooling is only in an orange path since a minimum of 5 CETs must be indicating > 1200 degrees for a red path condition per the background document, and the remaining CETs would only support an orange path condition, therefore FR-C.1 is not the correct procedure to implement at this time. The depressurization of the RCS to dump the accumulators would not be attempted until starting the SI pumps is attempted.

Examination:	RO 2008 Ginna NRC	**Question #:**	100	Rev. 2	Level: SRO
Lesson Plan:	RFRH1C	Response to Loss Of Secondary Heat Sink FR-H.1			
Objective(s):	2.01	Given a set of plant and equipment conditions, evaluate the conditions to determine the applicable procedure, and from the procedure determine the appropriate EXPECTED ACTIONS or RESPONSE NOT OBTAINED instructions to implement in FR-H.1, Response to Loss Of Secondary Heat Sink.			
Category:	Tier 3	**Question Source:**	Bank		
Cognitive level:	C/A	**Difficulty Level:**	3		
K/A:	2.4.16 Knowledge of EOP implementation hierarchy and coordination with other support procedures or guidelines such as, operating procedures, abnormal operating procedures, and severe accident management guidelines.	**ROI:**	3.5	**SROI:**	4.4
Technical References:	FR-H.1, Response to Loss Of Secondary Heat Sink pg. 9,15,16 F-0.2 and 3 (both pg. 2) Background document for F-0.2 pg. 5	**References provided during Exam:**	None		
Question History:	Used on Last (2) NRC Exams - No Used on Audit Exam - No	**Exam Bank #:**	PWR Industry Common Exam Bank		
10CFR 55 Content:	55.41	55.43(b)(5)			