Level SRO Tier 1 Group 1 K/A# 007 EA2.04 Imp. RO 4.6 Imp. SRO 4.4

1. Assume Unit 1 is operating at 8% power.

   a. If the reactor trip breakers do NOT automatically open when _____, then an ATWS is in progress.

   b. What actions should be performed for the ATWS event?

	EVENT	ACTIONS
a.	12 RCP shaft breaks with loop flow dropping to 0%	Remain in E-0 because reactor power is less than 10%. The E-0 immediate actions are: 1. Verify reactor trip 2. Verify turbine trip 3. Verify both safeguards buses – ENERGIZED 4. Check if SI is actuated.
b.	12 RCP shaft breaks with loop flow dropping to 0%	If a manual reactor trip did not work, then transition to FR-S.1 and 1. Manually insert control rods 2. Verify turbine trips.
c.	Intermediate Range N35 fails high	Remain in E-0 because reactor power is less than 10%. The E-0 immediate actions are: 1. Verify reactor trip 2. Verify turbine trip 3. Verify both safeguards buses – ENERGIZED 4. Check if SI is actuated.
d.	Intermediate Range N35 fails high	If a manual reactor trip did not work, then transition to FR-S.1 and 1. Manually insert control rods 2. Verify turbine trips.

ANSWER: D

Explanation:

A loss of RCS flow does not open the reactor trip breakers if <P-10 (i.e. 10% power). So this is not an ATWS event in the assumed condition. Since loss of flow is a potential reactor trip signal, this is a plausible distracter.
IR NI trip coincidence is ½ and would cause an automatic reactor trip because it is not blockable until >P-10.

Even if the reactor does not trip, we do not have to implement FR-S.1 if reactor power is less than 5% power. So the option to remain in E-0 is plausible if the immediate action power level is not remembered.

Technical References: **E-0 Step 1, FR-S.1 Steps 1, T.S. LCO 3.3.1**

Objective: **P8197L-014**

KA Statement: **Ability to determine and interpret the following as they apply to a reactor trip: If reactor should have tripped but has not done so, manually trip the reactor and carry out actions in ATWS EOP**

Cog. Level: **HIGH** **10CFR55.41:** **YES** **10CFR55.43:** **New Question:** **YES**

Level	SRO	Tier	1	Group	1	K/A#	011 EA2.02	Imp. RO	3.3	Imp. SRO	3.7

2. **Reference available: 2ES-1.2, "Transfer to Recirculation", Page 3 of 12.**

Assume all plant equipment was OPERABLE prior to the accident.
A large break loss of coolant accident has occurred on Unit 2. The Lead has just performed step 2 of 2ES-1.2, "Transfer to Recirculation."

The following indications were observed during performance of step 2:
1. The "LEFT" SI reset pushbutton was depressed. (Train A)
   - The "AUTOMATIC SI RESET" blue light remained dark. (47014:0504)
   - The "SI ACTUATED" blue light remained LIT. (47014:0604)
2. The "RIGHT" SI reset pushbutton was depressed. (Train B)
   - The "AUTOMATIC SI RESET" blue light illuminated. (47014:0504)
   - The "SI ACTUATED" blue light remained LIT. (47014:0604)

Step 4 reads "Check Both Trains Of Safeguards Pump(s) Available For Recirculation."
What is your assessment of the RHR system based on the indications provided?

a. Both trains of RHR are available for recirculation and 2ES-1.2 should work without the need to perform RNO actions.
b. Both trains of RHR are available for recirculation BUT the 21 RHR pump must be placed in PULLOUT during switchover.
c. Both trains of RHR are available for recirculation BUT the 22 RHR pump must be placed in PULLOUT during switchover.
d. Both trains of RHR are NOT available for recirculation and a transition should be made to 2ECA-1.1, "Loss of Emergency Coolant Recirculation".

ANSWER: B

Explanation:
Based on the indications provided, Train A SI did not reset. This affects the 21 RHR pump but not the associated valves which need to be positioned for the swapover to recirculation.
a Plausible if abnormal indications are not recognized.
b Correct.
c Plausible if indications are misunderstood or Train is mixed up.
d Plausible if indications are misunderstood but also added to balance options.

Technical References:
Westinghouse logics for SI
NF-40782-1, RHR Pump Logic
2ES-1.2, Transfer to Recirculation

Objective: **P8197L-012**

KA Statement: **Ability to determine and interpret the following as they apply to a Large Break LOCA: Consequences to RHR of not resetting safety injection**

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	1	Group	1	K/A#	027 2.4.30	Imp. RO	2.2	Imp. SRO	3.6

3. IF Unit 2 is in MODE 1 at 100% power, which set of conditions will require notification of the NRC within 1 hour per 10CFR50.72?

a. The "WHITE" pressurizer pressure transmitter (PT-430) fails off-scale high with the pressure control selector switch in the NORMAL position (WHITE-BLUE). RCS pressure drops to 2100 psig during the mitigating actions. A reactor trip does not occur.
b. Seat leakage of <0.5 gpm has been verified through pressurizer PORV CV-31231 by acoustic monitor and temperature indication. The leakage is still present with the associated block valve closed. PRT level is rising very slowly. RCS pressure is stable.
c. Failure of a pressurizer low level bistable has de-energized all of the pressurizer heaters. The heaters will not turn on in auto or manual control. RCS pressure remains above 2200 psig during the mitigating actions. The bistable is replaced and heaters energized in 55 minutes.
d. The PRZR PRESS MASTER CONTROL fails with a constant 90% output. The RO has taken manual control of pressurizer heaters and sprays. RCS pressure remained between 2200 and 2250 psig during the transient. Estimated time to return controller to automatic is 72 hours.

ANSWER: B

Explanation:

a Plausible because event caused entry into DNB LCO which seems important.
b Correct. These conditions constitute an Unusual Event per F3 and require prompt NRC notification. These conditions also require notification of the NRC resident per SWI-O-28.
c Plausible because these conditions require notification of the NRC resident per SWI-O-28 but incorrect because that is only a courtesy notification.
d Plausible since this event would disable automatic pressure control which seems important.

Technical References: F3-2, "Classifications of Emergencies"
T.S. LCO 3.4.9
SWI-O-28, "Notification of GSPO and NRC Resident Inspector"

Objective: P7410L-050

KA Statement: Emergency Procedures/Plan: Knowledge of which events related to system operations/status should be reported to outside agencies. (Pressurizer Pressure Control Malfunction)

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

The NRC reviewed this question and determined it should be deleted from the exam based on licensee post exam comments.

Level	SRO	Tier	1	Group	1	K/A#	054 AA2.01	Imp. RO	4.3	Imp. SRO	4.4

4. Unit 2 was operating normally at 100% power with all equipment OPERABLE and operating in the normal preferred lineups when both Main Feed Pumps tripped.
The following conditions now exist on Unit 2:
- RCS Tavg is 490°F and lowering
- 21 SG WR level is 15% and lowering
- 22 SG WR level is 35% and stable
- 21 MSIV is open and 22 MSIV is closed
- Both RCPs are stopped
- Both reactor trip breakers are closed.
- Both AFW pumps are running
- 60 seconds have elapsed since the trip of the feed pumps

Based on the plant indications, the _____(1)_____ has NOT tripped and accident analysis limits _____(2)_____ be violated.

	(1)	(2)
a.	turbine	may
b.	turbine	will not
c.	reactor	may
d.	reactor	will not

ANSWER: A

Explanation:

The distracters represent the possible choices in a 2 out of 2 twice question.

a Correct.

b Plausible because the turbine has NOT trip based on the change in RCS temperature and the SG level. Incorrect because the loss of normal feedwater ATWS accident analysis assumes AMSAC/DSS trips the main turbine when SG WR level is less than or equal to 40%. Since that assumption is not met, the accident analysis limits may be violated.

c Plausible because the reactor trip breakers are still closed. Incorrect because DSS has obviously completed the reactor scram.

d Plausible because the reactor trip breakers are still closed. Incorrect because DSS has obviously completed the reactor scram.

Technical References: **USAR Chapter 14, Section 14.8, ANTICIPATED TRANSIENT WITHOUT SCRAM (ATWS)**
Integrated plant knowledge

Objective: **P8174L-003, Objective 9c: Explain the importance of the system to plant safety: Accident analysis involving the system. (SRO only)**

KA Statement: **Ability to determine and interpret the following as they apply to the Loss of Main Feedwater (MFW): Occurrence of reactor and/or turbine trip**

Cog. Level:	**HIGH**	**10CFR55.41:**		**10CFR55.43:**	**YES**	**New Question:**	**YES**
Bank:		**Ques. ID:**		**Modified:**		**Last NRC Exam:**	

Level	SRO	Tier	1	Group	1	K/A#	055 EA2.02	Imp. RO	4.4	Imp. SRO	4.6

5. When directed to verify Natural Circulation (NC) flow during recovery from a loss of ALL AC power event, the following indications are present:
RCS pressure: 2235 psig and stable
11 SG: Level: 65% NR and lowering Pressure: 995 psig and slowly lowering
12 SG: Level: 60% NR and stable Pressure: 980 psig and slowly lowering
Core exit temperature: 539°F and stable
Loop A: Tcold: 517°F and lowering Thot: 539°F and stable
Loop B: Tcold: 505°F and stable Thot: 539°F and stable
RCS subcooling: 114°F and stable
Containment pressure is 0.1 psig

a. Is natural circulation established?
b. 1) IF established, then what action would promote an INCREASE in NC flow?
2) IF NOT established, then what action is needed to establish NC flow?

	NC ESTABLISHED	ACTION
a.	YES	Increase auxiliary feedwater flow to both SGs.
b.	YES	Manually open PORVs on both SGs.
c.	NO	Increase auxiliary feedwater flow to both SGs.
d.	NO	Manually open PORVs on both SGs.

ANSWER: D

Explanation: **The choices are all plausible because NC flow is either established or not established and both actions specified will promote NC flow. AFW flow should not be increased in the case given because SG levels are already higher than the normal control range specified in the EOPs. NC is not established because the SG are hotter than the RCS based on the values given. This makes only choice D correct.**

Technical References: **ECA-0.1 Attachment A, "Natural Circulation Conditions"**

Objective: **P8197L-011**

KA Statement: **Ability to determine and interpret the following as they apply to a Station Blackout: RCS core cooling through natural circulation cooling to S/G cooling**

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	NO
Bank:	INPO	Ques. ID:	#551 Beaver Valley	Modified:	YES	Last NRC Exam:	1997

Level	SRO	Tier	1	Group	1	K/A#	057 2.2.25	Imp. RO	2.5	Imp. SRO	3.7

6. **Reference available: T.S. LCO 3.8.9, "Distribution Systems – Operating."**

On June 10 at 1000, Unit 1 was in Mode 1 at 98% power.
Sequence of events:

- 06/10 @ 1000, 480v motor control center (MCC) 1X1 was declared inoperable due to a concern about the seismic mounting of the MCC. Unit 1 entered T.S. LCO 3.8.9.
- 06/10 @ 1800, A power reduction was started per Action D.
- 06/10 @ 1925, 480v MCC 1X2 was also declared inoperable.
- 06/10 @ 2355, Unit 1 entered Mode 3.
- 06/11 @ 0100, Unit 1 started a cooldown to Mode 5.
- 06/11 @ 0615, Arcing was observed in 120v AC panel 113. The panel was de-energized and a status evaluation was started.
- 06/11 @ 0730, 480v MCCs 1X1 and 1X2 were declared OPERABLE. Plant cooldown was stopped with Unit 1 at 390°F.

What action(s) is required for this event and when must the action(s) be accomplished?

a. The cooldown should resume immediately. Unit 1 must be in Mode 5 by 06/12/03 @ 0600. Cooldown may be stopped when Panel 113 is again OPERABLE.
b. The cooldown should resume immediately. Unit 1 must be in Mode 5 by 06/12/03 @ 1400. Cooldown may be stopped when Panel 113 is again OPERABLE.
c. The cooldown should resume immediately. Unit 1 must be in Mode 5 by 06/12/03 @ 1815. Cooldown may be stopped when Panel 113 is again OPERABLE.
d. Panel 113 must be returned to an OPERABLE status by 06/11/03 @ 0815 or Unit 1 must start a cooldown and be in Mode 5 by 06/12/03 @ 2015.

ANSWER: B

Explanation:

To answer this question the SRO must understand the completion time of 16 hours from entering the LCO which is explained in the bases section.

a **Plausible since this is the time when Mode 5 was required if Condition A was not meet. However, Condition A was exited and now Condition C is limiting and not met but an additional 8 hours are available.**

b **Correct. Action C is entered at 0615 on the 11th but the cumulative a completion time of 16 hours has already been exceeded. We must comply with the completion times of Action D.**

c **Plausible if the 36 hour completion time is indexed to entering Action C instead of not meeting the LCO.**

d **Plausible if Action C is referenced without considering the fact the LCO has not been meet since Action A was entered.**

Technical References: **T.S. LCO and Bases for 3.8.9.**

Objective: **P8186L-015**

KA Statement: **Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (Loss of Vital AC Instrument Bus)**

Cog. Level: **HIGH** **10CFR55.41:** **10CFR55.43:** **YES** **New Question:** **YES**

The NRC reviewed this question and determined the correct answer is A vice B based on licensee comments.

Level	SRO	Tier	1	Group	1	K/A#	E11 EA2.1	Imp. RO	3.5	Imp. SRO	4.2

7. Following a LOCA, low head recirculation flow was established on both RHR trains. Several hours later, debris blocks the suction lines from Containment Sump B to both trains of RHR and both RHR pumps are cavitating. The RO suggests making a transition to procedure ECA-1.1, "Loss of Emergency Coolant Recirculation"

   SHOULD a transition be made to ECA-1.1 and **WHY**?

a. No, ECA-1.1 strategies would NOT provide a success path for these conditions.
b. Yes, ECA-1.1 would still be effective in restoring long-term cooling.
c. Yes, ECA-1.1 would provide temporary core cooling until the containment is flooded.
d. No, instead start the SI pumps and return to E-1, "Loss of Reactor or Secondary Coolant."

ANSWER: A

Explanation:

a **Correct. Rules of usage do not allow entry to ECA-1.1 after a successful completion of the transfer to recirculation and actions in ECA-1.1 will not be effective with the RWST empty.**

b. **Plausible based on the title but not true because ECA-1.1 does not have actions to correct a suction problem.**

c **Plausible because the statement is true without condition. However, the procedure would not be used under the conditions given.**

d **Plausible because this action is on the information page of ES-1.1, Post-LOCA Cooldown and Depressurization if subcooling is lost but it is not appropriate because the SI pumps have no suction source.**

Technical References: **ECA-1.1**

Objective: **P8197L-012**

KA Statement: **Ability to determine and interpret the following as they apply to Loss of Emergency Coolant Recirculation: Facility conditions and selection of appropriate procedures during abnormal and emergency operations.**

Cog. Level:	**HIGH**	**10CFR55.41:**		**10CFR55.43:**	**YES**	**New Question:**	**YES**
Bank:		**Ques. ID:**		**Modified:**		**Last NRC Exam:**	

The NRC reviewed this question and deleted it from the examination based on the licensee's post-examination comments.

Level	SRO	Tier	1	Group	2	K/A#	024 AA2.01	Imp. RO	3.8	Imp. SRO	4.1

8. **Reference available: C12.5 AOP1, "Emergency Boration of the Reactor Coolant System."**

Due to a turbine load rejection, the control rods are currently about 10 steps below the rod insertion limit. The RO has attempted to establish the required boration flow using the normal boration flowpath but was unsuccessful due to apparent line blockage at the blender. The RO has just completed the actions of C12.5 AOP1, "Emergency Boration of the Reactor Coolant System" to establish emergency boration flow but has reported that there did NOT appear to be any flow in this flowpath either. The following conditions exist:

- Steps 1 through 2.4.3 of the AOP were performed with expected main control board indications.
- There is NO response on 1YIC113, EMERGENCY BORATION INTEGRATOR.
- RCS temperature is stable.
- Control rods are NOT moving.
- Step 2.4.3 was performed about 1 minute ago.

What action do you take?

a. Direct the RO to monitor alternate indications such as VCT level for evidence of boration.
b. Immediately enter T.S. LCO 3.0.3 and commence an orderly plant shutdown.
c. Borate using C18 AOP1, "Makeup or Boration of the RCS Using a Safety Injection Pump".
d. Reduce turbine load to allow manual rod withdrawal until rods are above the rod insertion limit.

ANSWER: A

Explanation:

a This is the correct answer. Counter malfunctions have occurred and SWI-O-10 states that "verify" means to take the appropriate contingency action whether stated or implied.

b Plausible because LCO 3.0.3 is sometimes entered in T.S. required actions can not be completed but in this case we have 1 hour to initiate the boron. NOTE: Old T.S. required the boration to be performed immediately.

c Plausible because of the title but incorrect because this procedure can NOT be entered unless the reactor is shutdown.

d Plausible because turbine load reduction is used to clear the RIL with a stuck rod but it would make the existing situation worse if boration is not available.

Technical References:
C12.5 AOP1, "Emergency Boration of the Reactor Coolant System"
C18 AOP1, "Makeup or Boration of the RCS Using a Safety Injection Pump".
T.S. LCO 3.1.6, Control Bank Insertion limits
SWI-O-10, Step 7.8.1 (last paragraph)

Objective: P8197L-014

KA Statement: Ability to determine and interpret the following as they apply to the Emergency Boration: Whether boron flow and/or MOVs are malfunctioning, from plant conditions

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	1	Group	2	K/A#	003 AA2.03	Imp. RO	3.6	Imp. SRO	3.8

9. **Refer to ERCS Printouts (Exhibits: SRO-1 through SRO-3).**

Unit 1 was operating at 25% power near the end of cycle when RCS temperature dropped for no apparent reason. Rods initially stepped out in AUTO but the operators placed rod control in MANUAL after verifying that a turbine runback was NOT in progress. The transient was not particularly severe and a manual reactor trip was NOT performed. After the transient, the plant parameters were as shown on the referenced printouts.

What is the MOST LIKELY cause of the transient?

a. Xenon oscillation
b. Dropped control rod
c. Power range NI failure
d. RCS boron change

ANSWER: B

Explanation:

- **a** Plausible because radial xenon oscillation can affect power range instruments but not correct because xenon oscillation of the magnitude consistent with the power range indications do not spontaneously occur.
- **b** Correct.
- **c** Plausible because one power range is reading much lower than the others but no correct because a low failure would not affect Tavg or rod control.
- **d** Plausible because RCS boron levels will affect Tavg and rod control but not correct because RCS boron changes will not cause a flux tilt.

Technical References: Simulator response
Alarm Response C47013:0203, NIS POWER RANGE CHANNEL DEVIATION

Objective: P8184L-005

KA Statement: Ability to determine and interpret the following as they apply to the Dropped Control Rod:
Dropped rod using in-core/ex-core instrumentation, in-core or loop temperature measurements

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	1	Group	2	K/A#	074 2.2.25	Imp. RO	2.5	Imp. SRO	3.7

10. What is the basis for T.S. LCO 3.3.3, “Event Monitoring Instrumentation” requirement to have four (4) core exit thermocouples (CETs) OPERABLE?

The CETs are required to be OPERABLE…

a. to help monitor core peaking factors.
b. for primary calorimetric monitoring.
c. to monitor core bypass flow.
d. for subcooling monitoring.

ANSWER: D

Explanation:
- **a Plausible since core temperature can be used to monitor certain peaking factors.**
- **b. Plausible because temperature measurement is needed for a primary calorimetric.**
- **c. Plausible because core bypass flow can be calculated using CET and Thot values.**
- **d. Correct per T.S. bases although the term “EOP” is not explicitly stated.**

Technical References: **T.S. Bases for LCO 3.3.3 Function 15.**

Objective: **P8184L-004**

KA Statement: **Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (Inadequate Core Cooling)**

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	1	Group	2	K/A#	E10 EA2.2	Imp. RO	3.4	Imp. SRO	3.9

11. **Reference available: ES-0.4, "Natural Circulation Cooldown With Steam Void In The Vessel."**

Unit 1 has tripped and both RCPs are stopped due to mechanical problems. A natural circulation cooldown is in progress but a steam void has formed in the reactor vessel. Circumstances are forcing us to continue the plant cooldown at an accelerated rate using the guidance of ES-0.4, "Natural Circulation Cooldown With Steam Void In The Vessel." Step 9 has been completed and you are currently "looping" through Steps 3 through 9.

The following conditions exist:

- RCS cooldown is in progress with a stable cooldown rate of 80°F/hour.
- RCS cold leg temperatures are 480°F and decreasing.
- RCS hot leg temperatures are 530°F and decreasing.
- Maximum core exit thermocouple temperature is 544°F.
- RCS pressure is 1200 psig and stable.
- Pressurizer heaters are OFF except for Groups A and B.
- Letdown is in service with all orifice isolation valves open.
- A portion of charging flow is going to pressurizer spray.
- RCP seal injection flow is 8 gpm to each RCP.
- Pressurizer level is 88% and stable.
- RVLIS full range level is 78% and decreasing.

What action should you implement as Unit 1 Shift Supervisor?

a. Turn on available pressurizer heaters.
b. Reduce steaming rate to drop the RCS cooldown rate to <25°F/hour
c. Reduce charging pump speed to reduce auxiliary spray flow.
d. Actuate safety injection and transition to E-0, "Reactor Trip or Safety Injection."

ANSWER: A

Explanation:

Provide each candidate with a full copy of ES-0.4 and the Steam Tables. The plant is in a procedure loop between step 3 and step 9. However step 5 is not satisfied.

a Correct per step 5 RNO.

b Plausible because this would reduce the contraction of the RCS fluid and probably raise pressurizer level and pressure with no other operator action. NOTE: The NC cooldown limit is usually 25°F/hour but not in this procedure.

c Plausible because this will raise pressure but it will lower pressurizer level which will lower pressure. Net effect is that this action will not be very effective. In addition, this is the wrong action because we need the water in the pressurizer to restore vessel level.

d Plausible if subcooling is calculated incorrectly.

Tech. Ref: **ES-0.4, "Natural Circulation Cooldown With Steam Void In The Vessel"**

Objective: **P8197L-012**

KA Statement: **Ability to determine and interpret the following as they apply to the Natural Circulation with Steam Void in Vessel with/without RVLIS: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments**

Cog. Level: **HIGH** **10CFR55.41:** **10CFR55.43:** **YES** **New Question:** **YES**

Level	SRO	Tier	1	Group	2	K/A#	E13 EA2.2	Imp. RO	3.0	Imp. SRO	3.4

12. **References available:**
    - **2FR-H.2, "Response to Steam Generator Overpressure"**
    - **2FR-H.3, "Response to Steam Generator High Level"**
    - **2ES-1.1, "Post-LOCA Cooldown and Depressurization" (minus info page)**

The following conditions exist:
- A small break LOCA has occurred on Unit 2.
- ES-1.1,"Post-LOCA Cooldown and Depressurization" is being implemented.
- Average core exit thermocouple temperature is 470°F.
- RCS pressure is 850 psig.
- RCS subcooling is 52°F.
- Both RCPs are operating.
- MSIVs are closed on both steam generators.
- Feedwater is isolated to BOTH steam generators.
- 21 SG has pressure at 1150 psig and narrow range level at 95% and stable.
- 22 SG has pressure at 520 psig and narrow range level at 54% and slowly lowering.
- The Heat Sink Critical Safety Function Status Tree is Yellow due to 21 SG pressure.
- In ES-1.1, we are about to perform step 5.

The Shift Manager has asked you to make a recommendation on actions to be performed.

You recommend…

a. remaining in ES-1.1 but NOT using 21 SG for the RCS cooldown in step 5.
b. transitioning EOP actions to FR-H.2, "Response to Steam Generator Overpressure".
c. transitioning EOP actions to FR-H.3, "Response to Steam Generator High Level."
d. transitioning EOP actions to E-3, "Steam Generator Tube Rupture".

ANSWER: A

Explanation:	a	**Correct as noted in FR-H.2 steps 2 & 6 and the caution before step 4.**
	b	**Plausible because entry conditions are met but incorrect because we do not implement YELLOW path FRPs over ES procedures.**
	c	**Plausible because that is the wording on Step 3 RNO but not appropriate since we do NOT implement Yellow path FRPs over ES procedures.**
	d	**Plausible because the information page of ES-1.1 has a E-3 transition based on SG level but incorrect because SG high level isn't due to SGTR since RCS pressure < SG pressure.**
Technical References:	**FR-H.2, Response to Steam Generator Overpressure** **ES-1.1, Post LOCA Cooldown and Depressurization** **1F-0, Critical Safety Function Trees Background**	
Objective:	**P8197L-014**	
KA Statement:	**Ability to determine and interpret the following as they apply to the Steam Generator Overpressure: Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments**	

Cog. Level: **HIGH** **10CFR55.41:** **10CFR55.43:** **YES** **New Question:** **YES**

Level	SRO	Tier	2	Group	1	K/A#	012 2.2.25	Imp. RO	2.5	Imp. SRO	3.7

13. **References available: T.S. LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation"**

For corrective maintenance, power was lowered to 8% on Unit 1. In this condition, I&C had to perform SP-1198, "NIS Power Range Startup Test." At the end of the surveillance test on channel N42, the I&C Specialist informed you that the power range low setpoint reactor trip function on channel N42 was found set at 37% power. The I&C Specialist also told you the trip function setpoint could be lowered no further than 25.2% which was NOT within the desired range of 23.9% to 24.9% in the SP. The as left setpoint was recorded as 25.2%.

What was and what is the status of the power range low power reactor trip function?

	AS FOUND	AS LEFT
a.	OPERABLE	OPERABLE
b.	Inoperable	OPERABLE
c.	OPERABLE	Inoperable
d.	Inoperable	Inoperable

ANSWER: C

Explanation:

Distracters represent a balanced set of two options available.

- **a** Plausible because the "as found" setpoint is acceptable per the T.S. Allowable Value.
- **b** Plausible because the "As found" is outside the "desired" range of the SP.
- **c** Correct per T.S. bases 3.3.2 page B3.3.2-6. Note: Information is not as clear in T.S. 3.3.1 bases but same intent should be assumed.
- **d** Plausible because the "As found" and "As left" are outside the "desired" range of the SP.

Technical References: T.S. LCO 3.3.1 and 3.3.2 Bases (discussion of allowable values)

Objective: P8184L-004

KA Statement: Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (RPS)

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	2	Group	1	K/A#	013 2.2.25	Imp. RO	2.5	Imp. SRO	3.7

14. The Engineered Safety Feature Actuation System (ESFAS) automatic logic for AFW actuation is required to be OPERABLE in MODES 1, 2 and 3. However, the auto-start from undervoltage (UV) on the associated 4KV buses is only required to be OPERABLE in MODES 1 and 2. Why?

a. In MODE 3, the thermal power is limited to decay heat only so the UV auto-start is not needed.
b. RCP UV trip is blocked below P-7 and this auto-start exists to promote NC when the RCPs trip.
c. This auto-start anticipates the loss of both MFW pumps which are not required in MODE 3.
d. The UV start is associated with the TDAFW pump which is NOT required in MODE 3.

ANSWER: C

Explanation:
- **a** Plausible because hypothesis is true but the conclusion is NOT related to the question.
- **b** Plausible because hypothesis is true but the conclusion is NOT related to the question.
- **c** Correct per T.S. LCO 3.3.2 bases.
- **d** Plausible because hypothesis is true but the conclusion is false.

Technical References: T.S. LCO 3.3.2 Bases

Objective: P8180L-006

KA Statement: Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (ESFAS)

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	2	Group	1	K/A#	062 2.4.6	Imp. RO	3.1	Imp. SRO	4.0

15. IF Unit 1 loses all AC power, ECA-0.0 will attempt to restore AC power to at least one Safeguard Bus. IF all possible sources are available, what is the PREFERRED BUS and POWER SUPPLY for power restoration?

a. Bus 15 supplied by D1.
b. Bus 15 supplied by 1RY.
c. Bus 16 supplied by D2.
d. Bus 16 supplied by 1RY.

ANSWER: D

Explanation:	**ECA-0.0 step 5 background explains that Bus 16 should be energized first because of the MDAFW pump but this is changing because the presence of 2 charging pumps is more critical to the strategy for an SBO event. In the ECA-0.0 revision in progress, the basis will be changed and there will be a unit difference at this step. The offsite supplies are preferred over the DG because of the greater load capacity associated with the offsite supplies.** **All distracters are plausible but do not match the mitigation order identified by step sequencing.**
Technical References:	**ECA-0.0 and associated background.**
Objective:	**P8197L-014**
KA Statement:	**Emergency Procedures/Plan: Knowledge symptom based EOP mitigation strategies. (AC Electrical Distribution)**

Cog. Level:	LOW	10CFR55.41:	YES	10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	2	Group	1	K/A#	064 2.2.22	Imp. RO	3.4	Imp. SRO	4.1

16. **References available:**
    - **T.S. LCO 3.8.1, "AC Sources - Operating"**
    - **5AWI 3.13.5, "Operability Determinations"**

An operator investigating a D6 local alarm reports that the relief valve for 1A Starting Air Receiver has failed open. The control room directs the operator to isolate the 1A air receiver from D6.

What is the status of D6?

D6 is…

a. OPERABLE
b. OPERABLE but degraded
c. Inoperable but available (for PRA purposes)
d. Inoperable and not available (for PRA purposes)

ANSWER: A

Explanation:	**Per C20.7, "D5/D6 Diesel Generators", Limitation 4.4, any one of our Air Start Systems may be OOS indefinitely. AWI 3.15.5 defines "degraded" as a condition of an SSC in which there has been a loss of quality or functional capability. If a degraded condition exists, we are required to develop an action plan to address the condition. Allowing a degraded condition to exist "indefinitely" is not allowed, so based on the information in 2C20.7, the condition described in the stem is NOT degraded.** **All distracters are plausible choices for an operability determination.**
Technical References:	**2C20.7, D5/D6 Diesel Generators** **H24.1, Assessment & Management of Risk Associated with Maintenance Activities**
Objective:	**P8186L-004**
KA Statement:	**Equipment Control: Knowledge of limiting conditions for operations and safety limits. (Emergency Diesel Generator)**

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	2	Group	2	K/A#	011 2.4.4	Imp. RO	4.0	Imp. SRO	4.3

17. Pressurizer level is used as a decision point for SI actuation or SI re-initiation in several procedures. In which circumstance should we **NOT** initiate Safety Injection flow and transition to a different procedure?

a. Mode 1, implementing 1C4 AOP2, "Steam Generator Tube Leak", pressurizer level 30% and lowering with maximum charging.
b. Mode 3, implementing 1FR-P.1, "Response to Imminent Pressurizer Thermal Shock", SI terminated, pressurizer level 4% and lowering.
c. Mode 3, implementing 1ES-0.1, "Reactor Trip Response", pressurizer level is 3% and stable.
d. Mode 4, implementing 1ES-1.1, "Post-LOCA Cooldown and Depressurization", SI terminated, pressurizer level 6% and lowering.

ANSWER: B

Explanation:
- **a** For SG tube leak, there is no level value to initiate SI but reactor trip and SI are actuated if level can not be maintained with available charging flow.
- **b** Correct. FR-P.1 only uses RVLIS level to re-initiate SI.
- **c** Post trip, we should initiate SI if pressurizer level can not be maintained above 5%.
- **d** Post-LOCA, SI reinitiation pressurizer level criterion is 7%.

Technical References: 1C4 AOP2, "Steam Generator Tube Leak", 1FR-P.1, "Response to Imminent Pressurizer Thermal Shock", 1ES-0.1, "Reactor Trip Response", 1ES-1.1, "Post-LOCA Cooldown and Depressurization",

Objective: P8197L-011, P8197L-012, P8197L-014

KA Statement: Emergency Procedures/Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (Pressurizer Level Control System)

Cog. Level:	LOW	10CFR55.41:	YES	10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

The NRC reviewed this question and determined that both answers B and D are correct based on licensee post exam comments.

Level	SRO	Tier	2	Group	2	K/A#	028 2.2.25	Imp. RO	2.5	Imp. SRO	3.7

18. Following a LOCA, both Hydrogen Recombiners were placed in service. For the next 4 hours, the containment hydrogen concentration continued to rise but at a slower rate.

Now, we are still implementing ES-1.1, "Post-LOCA Cooldown and Depressurization." Containment hydrogen concentration is 0.9% but the Train B hydrogen recombiner has lost power. The electricians believe there is a cable fault inside the containment such that the recombiner can NOT be returned to service.

What is the expected trend of the hydrogen concentration in containment?

a. Hydrogen concentration will remain below the 4% using only the Train A recombiner.
b. Hydrogen concentration will peak above the 4% but remain below the 6% flammability limit.
c. IF containment spray is NOT placed in service, THEN hydrogen concentration will exceed the 6% flammability limit.
d. Without using POST-LOCA hydrogen venting, hydrogen concentration will exceed the 6% flammability limit.

ANSWER: A

Explanation: **Basis for T.S. LCO 3.6.7, "Hydrogen Recombiners" states that containment hydrogen concentration can be maintained below 4% with only one recombiner placed in service. b,c and d are plausible sounding but none are correct because the LOCA is progressing as expected based on still being in an "ES" procedure instead of an "ECA" or "FR" procedure.**

This is keyed HIGH cognitive because we are asking candidates to make a prediction based on current conditions and system interactions.

Technical References: **T.S. LOC 3.6.7 Basis**

Objective: **P8180L-008**

KA Statement: **Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (Hydrogen Recombiner and Purge Control System)**

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	NO
Bank:	INPO	Ques. ID:	#3285 Beaver Valley	Modified:	YES	Last NRC Exam:	1998

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.1.34	Imp. RO	2.3	Imp. SRO	2.9

19. The Chemist reports that the lithium concentration in the RCS is approaching its upper limit. What is the preferred action that should be taken to restore the lithium concentration to its proper level?

a. Letdown flow should be maximized to accelerate the cleanup using the CVCS mixed beds.
b. Letdown flow should be minimized to improve the DF of the CVCS mixed beds.
c. Letdown flow should be passed through the CVCS cation bed.
d. Letdown flow should be diverted to the HUT to establish a bleed and feed for the RCS.

ANSWER: C

Explanation:

The most probable reason is an accumulation of lithium associated with neutron capture by the soluble boron. This lithium accumulation drives RCS pH high and must be reduced to keep pH in the acceptable range.

- **a** Plausible because flowrate will accelerate cleanup but incorrect because the mixed bed is lithium saturated.
- **b** Plausible because slower flowrate will improve DF but incorrect because the mixed bed is lithium saturated.
- **c** Correct. This is a frequently performed activity early in core life.
- **d** This activity will reduce lithium but will cause other problems and is not used unless cation bed is not available.

Technical References: C12.2,"Purification and Chemical Addition…"

Objective:

KA Statement: Conduct of Operations: Ability to maintain primary and secondary plant chemistry within allowable limits.

Cog. Level:	HIGH	10CFR55.41:	YES	10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.1.6	Imp. RO	2.1	Imp. SRO	4.3

20. You are the Containment SRO during fuel handling when a fuel assembly is dropped on top of other fuel assemblies in the core. You note visible damage to the dropped fuel assembly and bubbles rising to the surface. What is your FIRST action?

a. Notify the control room of the conditions.
b. Actuate the containment evacuation alarm.
c. Use the plant page to direct containment evacuation.
d. Direct refueling personnel to don respirators.

ANSWER: A

Explanation:

a **Correct. Per SWI-O-10, the operators must know the immediate actions of C1.6 AOP1 and D5.2 AOP1. D5.2 AOP1 has 1 immediate action to initiate containment evacuation per C1.6 AOP1. C1.6 AOP1 has 2 immediate actions but the 1st is to notify the control room.**

b **Plausible because it is the 2nd immediate action of C1.6 AOP1.**

c **Plausible because it s an early subsequent action and seems to be the immediate action in D5.2 AOP1 but it is not.**

d **Plausible because these would be the first people affected by the rising gas bubbles.**

Technical References: **D5.2 AOP1, "Damage Fuel Assembly"**
C1.6 AOP1, "Containment Evacuation"

Objective:

KA Statement: **Conduct of Operations: Ability to supervise and assume a management role during plant transients and upset conditions.**

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.2.25	Imp. RO	2.5	Imp. SRO	3.7

21. LCO 3.0.3 states: "When an LCO is not met and the associated ACTIONS are not met, an associated ACTION is not provided, or if directed by the associated ACTIONS, the unit shall be placed in a MODE or other specified condition in which the LCO is not applicable. Action shall be initiated within 1 hour to place the unit, as applicable, in:
    - MODE 3 within 7 hours;
    - MODE 4 within 13 hours; and
    - MODE 5 within 37 hours.

    If we enter LCO 3.0.3 while already in MODE 3, how long do we have to enter MODE 5 from the time we declared our entry into the LCO?

a. 30 hours
b. 31 hours
c. 37 hours
d. 38 hours

ANSWER: C

Explanation:
The basis for LCO 3.0.3 says the time limit for reaching MODE 5 is not reduced. This question is more valid at this time because of the recent adoption of Improved Technical Specifications.
a Plausible if the 7 hours to reach MODE 3 is subtracted from the time to reach MODE 5.
b Plausible if 1 hour to take action is added to answer in distracter A.
c Correct based of T.S. LCO 3.0.3 basis discussion of completion time.
d Plausible if 1 hour to take action is added to the limit to reach MODE 5.

Technical References: **Discussion of completion times in T.S. section 1.3 (see example on page 1.3-4)**
Bases document for T.S. LCO 3.0.3

Objective:

KA Statement: **Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits.**

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.2.32	Imp. RO	2.3	Imp. SRO	3.3

22. IF the core loading pattern will be CHANGED during the next refueling outage to place the new fuel assemblies more toward the center of the core and the "twice-burned" assemblies more toward the periphery, THEN what affect would this loading pattern have on the unit?

Assume Kexcess will be the same at the beginning of both fuel cycles.

a. The expected full power loop ΔT value should be significantly lower for this fuel cycle when compared to the value of full power loop ΔT for the previous fuel cycle.
b. The expected full power loop ΔT value should be significantly higher for this fuel cycle when compared to the value of full power loop ΔT for the previous fuel cycle.
c. IF PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly below actual power when the 1st calorimetric is performed after the refueling outage.
d. IF PR NI channel gains are NOT changed during the outage, the PR NIs would read significantly above actual power when the 1st calorimetric is performed after the refueling outage.

ANSWER: C

Explanation:

a Plausible because ΔT may change each core load but this is not associated with loading pattern. ΔT changes are usually result of SG tube plugging or RCP changes.
b Plausible because ΔT may change each core load but this is not associated with loading pattern. ΔT changes are usually result of SG tube plugging or RCP changes.
c Correct based on many actual plant events.
d Plausible because core loading pattern affects NIs but incorrect because this change should make NIs read non-conservatively.

Technical References: IN 83-43, Improper Setting of High Flux Trip Setpoints
Objective:
KA Statement: Equipment Control: Knowledge of the effects of alterations on core configuration.

Cog. Level:	HIGH	10CFR55.41:		10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.3.3	Imp. RO	1.8	Imp. SRO	2.9

23. Given the following plant conditions:
    - Unit 1 Steam Generator Blowdown flow is being discharged to the river.
    - Radiation Monitor 1R-19 has just lost power.

    Which of the following actions should be taken?

a. Terminate discharge flow or obtain periodic effluent grab samples.
b. Reset blowdown in the Auxiliary Building and reopen the blowdown control valves.
c. IF R-18 discharge line monitor is operable, discharge may be resumed.
d. Terminate discharge flow because discharge is NEVER allowed with 1R-19 out of service.

ANSWER: A

Explanation:

a **Correct, sampling is now required or this would constitute an unmonitored release that must be terminated.**
b **Incorrect, blowdown should not be established without process monitoring.**
c **Incorrect, the S/G blowdown flash tank enters the common discharge header downstream of R-18 making R-18 ineffective in monitoring a continued discharge.**
d **Incorrect, discharge can be continued with appropriate sampling.**

CFRBASIS - 10CFR55.43(b)(4) Radiation hazards that may arise during normal and abnormal situations.

Technical References: **C21.1.3.2, Steam Generator Blowdown – Limitation 4.1**

Objective:

KA Statement: **Radiation Controls: Knowledge of SRO responsibilities for auxiliary systems that are outside the control room (e.g., waste disposal and handling systems).**

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	NO
Bank:	INITIAL1	Ques. ID:	P8182L-001A (005)	Modified:	NO	Last NRC Exam:	----

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.3.4	Imp. RO	2.5	Imp. SRO	3.1

24. Given the following conditions on Unit 2:
    - A LOCA outside containment occurred at 0130
    - A Site Area Emergency was declared at 0140
    - The broken line was manually isolated locally, but the operator performing the task was injured and cannot leave the area on his own
    - Initial dose estimates are 90 R/hr gamma
    - The rescue time for a 2-man team is estimated to be 10 minutes with a maximum of 15 minutes

    Under these circumstances, a rescue attempt…

a. is **NOT** allowed because whole body exposure would exceed the emergency limit.
b. may be made by qualified individuals selected and approved by the Shift Supervisor.
c. may be made by anyone since 10CRF20 exposure limits will **NOT** be exceeded.
d. by risk-informed volunteers may proceed **ONLY** with Emergency Director authorization.

ANSWER: D

Explanation:
Section 4.0 of F3-12 states that the Emergency Director is the sole individual having authority to authorize exceeding 10CFR20 exposure limits. Precaution 3.7 states the 25 rem TEDE may be exceeded only by volunteers who are fully aware of the risks. Precaution 3.8 recommends using volunteers above the age of 45.
a Plausible if 25 REM limit is not known.
b Plausible if emergency exposure controls are not known.
c Plausible if 10CFR20 limits are not known.
d Correct per explanation above.

Technical References: **F3-12, "Emergency Exposure Control"**

Objective:

KA Statement: **Radiation Controls: Knowledge of radiation exposure limits and contamination control, including permissible levels in excess of those authorized.**

Cog. Level:	LOW	10CFR55.41:		10CFR55.43:	YES	New Question:	NO
Bank:	INITIAL1	Ques. ID:	P9130L-003 (041)	Modified:	YES	Last NRC Exam:	----

Level	SRO	Tier	3	Group	1	K/A#	GEN 2.4.27	Imp. RO	3.0	Imp. SRO	3.5

25. If there is a fire in the plant, the Shift Manager shall proceed to the…

a. scene and assume the role of Fire Brigade Chief.
b. scene to determine if the Red Wing Fire Department should be called.
c. control room and assume responsibility for coordinating fire fighting activities.
d. technical support center and assume Emergency Director responsibilities.

ANSWER: C

Explanation:
- **a** Plausible based on desire to have senior personnel in charge.
- **b** Plausible based on common sense.
- **c** Correct based on Step 2.3 of procedure F5.
- **d** Plausible but E-plan is not always activated for a fire. (i.e. < 15 minutes)

Technical References: F5, Fire Fighting"

Objective:

KA Statement: Emergency Procedures/Plan: Knowledge of fire in the plant procedure.

Cog. Level:	LOW	10CFR55.41:	YES	10CFR55.43:	YES	New Question:	YES
Bank:		Ques. ID:		Modified:		Last NRC Exam:	