Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 1	Group #		1		
	K/A #		007 Reactor Trip-Stabilization EK2 Knowledge of the interrelationship between the reactor trip and the following: EK2.02 Breakers, Relays and Disconnects		
	Importance Rating		2.6		

Proposed Question:

The following plant conditions exist:

- The reactor has just tripped from 100% power.
- 'B' Reactor Trip Breaker (RTB) is CLOSED.
- 'A' Reactor Trip Breaker (RTA) is OPEN.
- The steam dumps are in the Tavg mode.

Which of the following describes the automatic operation of the Steam Dumps as a result of this transient?

A. The Steam Dumps will OPEN on the Plant Trip Controller.

B. The Steam Dumps will OPEN on the Load Rejection Controller.

C. The Steam Dumps will remain CLOSED until the Steam Pressure Mode is selected.

D. The Steam Dumps will remain CLOSED until the 'B' Reactor Trip Breaker is opened.

Proposed Answer: B

B is correct. The steam dumps will arm based on a Train 'A' P-4 signal, however the steam dump controller will not swap to the Plant Trip Controller as this requires a Train 'B' P-4 signal. There is no Train 'B' P-4 signal due to failure of the 'B' Reactor trip breaker to open.

A is incorrect but plausible. The steam dumps will open, however they would not be operating with the Plant Trip Controller because the Train 'B' P-4 signal is not present.

C is incorrect but plausible. The Train 'A' and Train 'B' Reactor Trip Breakers serve to arm the steam dumps and swap controllers. The student must know that Train 'A' P4 arms the Steam Dumps. If the student decides that Train 'B' P-4 arms the steam dumps then they may believe that the controller would have to be swapped to the Steam Pressure Mode.

D is incorrect but plausible. If the candidates confuse the function of the Train 'A" and Train "B' P-4 signals then they may think that the steam dumps would not arm and function until the 'B' Reactor Trip Breaker was opened.

Technical Reference(s):	1-NHY-509042, Reactor Trip Signals w/ Functional diagrams		1-NHY-509050, MS Dump Control w/ Functional Diagrams

Proposed references to be provided to applicants during examination:	None
K/A Topic:	007 Reactor Trip-Stabilization
Question Source:	Bank. Teb 20902
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.7
Learning Objective:	L8056I21

ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 2	Group #		1		
	K/A #		009 Small Break LOCA EA2 Ability to determine or interpret the following as they apply to the small break LOCA: EA2.37 Existence of adequate natural circulation.		
	Importance Rating		4.2		

Proposed Question:

Given the following plant conditions:

- A LOCA has occurred.
- All Reactor Coolant Pumps have been tripped.
- The crew is performing procedure ES-1.2, Post LOCA Cooldown and Depressurization."
- Reactor Coolant System pressure is 1490 psig and stable.
- All Wide Range T_{cold} temperatures are 508°F and slowly decreasing.
- All wide Range T_{hot} temperatures are 579°F and slowly decreasing.
- Core Exit Thermocouple temperatures are 581°F and stable.
- Steam Generator pressures are 715 psig and slowly decreasing.
- Steam Generator narrow range levels are being maintained at approximately 30%.
- Containment pressure is 2 psig and stable.

What is the status of natural circulation flow in the Reactor Coolant System?

A. Natural circulation cannot be verified because there is inadequate subcooling.

B. Natural circulation cannot be verified because Steam Generator conditions are not satisfied.

C. Natural circulation cannot be verified because Core Exit Thermocouple temperature is not decreasing.

D. All conditions indicate that natural circulation flow has been established.

Proposed Answer:	A	

A is correct. Per ES-1.2, Post LOCA Cooldown and Depressurization, Attachment G, the following conditions support or indicate natural circulation flow:

- RCS subcooling- Greater than 40°F
- Steam Generator pressures-Stable or Decreasing
- RCS Hot Leg temperatures-Stable or Decreasing
- Core Exit Thermocouples-Stable or Decreasing
- RCS Cold Leg temperatures- At Saturation Temperature for SG Pressure

RCS subcooling is 15.62°F based on RCS pressure and Core Exit Thermocouple temperatures. The RCS subcooling condition for natural circulation is not met. All other natural circulation criteria are met.

B is incorrect but plausible. The candidate may analyze secondary heat sink conditions based on SG level, however this is not a condition used to determine the status of natural circulation flow.

C is incorrect but plausible. It is plausible that the required condition must be Core Exit thermocouple temperatures decreasing, however the condition is satisfied if CETC temperature is stable. Even if this condition is met there is inadequate subcooling so natural circ conditions are not met

D is incorrect but plausible. It is plausible that all conditions would be satisfied if RCS subcooling criteria were based on RCS pressure vs. RCS loop wide range temperatures however RCS subcooling is calculated based on Core Exit Thermocouple temperature.

Technical Reference(s):	ES-1.2, Post LOCA Cooldown and Depressurization, Attachment G.		

Proposed references to be provided to applicants during examination:	Steam Tables
K/A Topic:	009 Small Break LOCA
Question Source:	Bank. Teb 22270
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	43.5/45.13
Learning Objective:	L1204I03, L1225I08

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 3	Group #		1		
	K/A #		011 Large Break LOCA EK3 Knowledge of the reasons for the following responses as they apply to the large break LOCA: EK3.14 RCP tripping requirement.		
	Importance Rating		4.1		

Proposed Question:	

Given the following plant conditions:

- A Loss of Coolant Accident (LOCA) has occurred.
- The reactor has tripped and Safety Injection has actuated.
- All ESF systems function as designed.
- E-O, "Reactor Trip or Safety Injection" is being implemented and the crew has just completed immediate action steps 1-4.
- Power is lost to 4.16kV Bus E-5.
- Containment pressure is 5 psig and increasing.
- Core Exit Thermocouple temperature is 544°F and decreasing.
- RCS pressure is 1270 psig and decreasing.

What is the required status of the Reactor Coolant Pumps for these current plant conditions and why?

A. One pump should be stopped to save it for use during future EOP strategies.

B. All of the pumps should be stopped because PCCW cooling flow to the RCP's has been lost.

C. All of the pumps should be stopped because inadequate RCS subcooling margin exists.

D. All pumps should remain running because there is inadequate high head safety injection capability.

Proposed Answer:	C	

C is correct. Procedure E-0 contains an OAS item that directs tripping all reactor coolant pumps if RCS subcooling is below 40°F and there is at least one CCP or SI pump running. The stem of the question states that Bus E-5 is deenergized however Bus E-6 is still energized so there is one CCP and one SI pump that should be running. Given the RCS temperature and pressure conditions listed in the question stem, subcooling is less than 40°F and all reactor coolant pumps should be stopped. Tsat for 1270 psig is 575°F. Actual RCS Temp is 544°F. Subcooling is 31°F.

A is incorrect but plausible. It is plausible that one pump should be saved for future use as the pumps may be physically challenged due to potential voided conditions in the RCS. The strategy of securing one RCP for future use is addressed in the Westinghouse EOP Background Document, Generic Issue, RCP Trip/Restart, however this strategy is utilized in degraded core cooling functional restoration procedures. The question stem states that the crew is still in procedure E-0 where critical safety functions and implementation of any functional restoration procedures is not yet in effect.

B is incorrect but plausible. Procedure E-0 contains an OAS item which directs tripping all reactor coolant pumps if PCCW cooling is lost, which would occur on a Phase B isolation signal (18 psig). The question stem states that containment pressure is at 5 psig which is below the Phase B isolation setpoint, however, with no electrical power to Bus E-5 PCCW cooling flow is lost to 2 of 4 pumps. This would eventually require shutdown of the two affected RCP's.

D is incorrect but plausible. Procedure E-0 does dictate keeping the RCP's running if there is not at least one CCP or SI pump running, however the stem of the question states that Bus E-5 is deenergized however Bus E-6 is still energized so there is one CCP and one SI pump that should be running.

Technical Reference(s):	E-0, Reactor Trip or Safety Injection, Operator Action Summary Page, RCP Trip Criteria (LOCA)		Westinghouse EOP Background Document, Generic Issue, RCP Trip/Restart

Proposed references to be provided to applicants during examination:	Steam Tables
K/A Topic:	011 Large Break LOCA
Question Source:	Modified from bank. Seabrook 2005 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.13
Learning Objective:	L1202I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 4	Group #		1		
	K/A #		026 Loss of Component Cooling Water AK3 Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: AK3.02 The automatic actions (alignments) within the CCWS resulting from actuation of ESFAS.		
	Importance Rating		3.6		

Proposed Question:

Given the following conditions:

- The plant was initially at 100% power.
- A LOCA occurred.
- Containment pressure increased to 19 psig.
- All ESFAS features actuated as designed.

What PCCW equipment alignment occurred and why?

A. The Train ‘A’ and Train ‘B’ Thermal Barrier Cooling System Isolation Valves were isolated by a “T” signal to further increase the cooling to safety related components.

B. The Train ‘A’ and Train ‘B’ PCCW Containment Isolation Valves were isolated by a “P” signal in order to minimize the atmospheric release of radioactive materials from containment.

C. The PCCW Outlet Valves from the Containment Building Spray Heat Exchangers opened by a “T” signal to supply containment building cooling.

D. The PCCW Outlet Valves from the RHR Heat Exchangers opened by a “P” signal to supply ECCS cooling.

Proposed Answer: B

B is correct. The Train B and Train B PCCW Containment Isolation Valves close on a Phase B (P) signal pursuant to the design basis of minimizing the release of radioactive materials in the event of a LOCA.

A is incorrect but plausible. A common operator misconception is that the PCCW containment isolation valves to the Thermal Barrier Cooling Water System are isolated by a ‘T’ signal.

C is incorrect but plausible. The PCCW Outlet Valves from the Containment Building Spray Heat Exchangers do automatically open however it is by a “P” signal.

D is incorrect but plausible. The PCCW Outlet Valves from the RHR Heat Exchangers do automatically open however it is by a “T” signal

Technical Reference(s):	UFSAR, Section 6.2.4.1, Containment Systems, Design Basis, item a.		UFSAR, Section 9.2.2.3, Water Systems, PCCW Safety Evaluation
	1-NHY-503268, Containment Structure PCCW Isolation Valves Logic Diagram		1-NHY-503273, CC Non Essential PCCW Isolation Valves Logic Diagram
Proposed references to be provided to applicants during examination:	None		
K/A Topic:	026 Loss of Component Cooling Water		
Question Source:	Modified. Braidwood 2007 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7/45.7		
Learning Objective:	L8036I12, L8057I08, L8057I10,		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 5	Group #		1		
	K/A #		027 Pressurizer Pressure Control System Malfunction 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretations.		
	Importance Rating		4.4		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- Pressurizer pressure is 2235 psig.
- Pressurizer pressure control is in automatic.
- The controller setpoint drifts from 2235 psig to 2160 psig and system components respond accordingly.
- The Master Pressure Controller is placed in MANUAL.

What action should be taken next?

A. Depress the INCREASE pushbutton. This will close both spray valves and energize the backup heaters.

B. Depress the DECREASE pushbutton. This will close both spray valves and energize the backup heaters.

C. Depress the INCREASE pushbutton. This will de-energize the backup heaters and open both spray valves.

D. Depress the DECREASE pushbutton. This will de-energize the backup heaters and open both spray valves.

Proposed Answer:	B

B is correct. When the controller setpoint drifts down from 2235 psig to 2160 psig then an error signal is generated based on actual pressure being greater than setpoint. This error signal would cause the controller output signal to increase causing heaters to deenergize and spray valves to open. The correct response is to close the spray valves and energize heaters by depressing the DECREASE pushbutton.

A is incorrect but plausible. When the controller setpoint drifts down from 2235 psig to 2160 psig then an error signal is generated based on actual pressure being greater than setpoint. This error signal would cause the controller output signal to increase causing heaters to deenergize and spray valves to open. The correct response is to close the spray valves and energize heaters, however depressing the INCREASE pushbutton would not be the appropriate action. This choice is plausible as there is a need to increase pressurizer pressure. The method of increasing pressurizer pressure is to decrease the controller output signal vice increase. ***There is a common operator misconception with regard the Pressurizer Pressure Controller based on the controller response dynamics due to the controllers integrating function and the associated component responses based on controller output signals.***

C is incorrect but plausible. If the student misinterprets the 2160 psig value as being the driving input signal to the controller then they would interpret that all heaters would be energized and spray valves would be closed. In this situation the student would interpret that pressure would continue to increase and there would be a need to depress the INCREASE pushbutton to stop the transient.

D is incorrect but plausible. If the student misinterprets the 2160 psig setpoint value as the driving process value then they would interpret that there would be a would a demand for heaters and spray valves would be closed. Depressing the DECREASE pushbutton would not be appropriate for this condition as the action would continue to call for a demand for heaters and closed spray valves.

Technical Reference(s):	1-NHY-509026, Pressurizer Pressure Control		Main Plant Computer System, Primary Tech Data, Pressurizer Controller Output

Proposed references to be provided to applicants during examination:	None
K/A Topic:	027 Pressurizer Pressure Control System Malfunction
Question Source:	Bank. Teb 26879
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.12/ 45.13
Learning Objective:	L8027I06, L8027I05, L8027I08, L8027I10

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 6	Group #		1		
	K/A #		029 ATWS EA1 Ability to operate and monitor the following as they apply to the ATWS: EA1.14 Driving of Control Rods into the core.		
	Importance Rating		4.2		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- All systems are aligned normally.
- Control rods are in MANUAL.
- The main turbine has tripped due to high bearing vibrations.
- A valid reactor trip signal is received and the reactor did NOT automatically trip.
- The Control Room Operator could not manually trip the reactor from the Main Control Board.
- The crew has entered FR-S.1, "Response to Nuclear Power Generation/ATWS."

What is the first action that should be taken in order to insert negative reactivity into the core?

A. Verify control rods are being inserted in auto OR manually insert control rods.
B. Close the Main Steam Isolation Valves and allow the RCS to heat up.
C. Align Charging Pump suction to the RWST and isolate suction from the VCT.
D. Start at least one Boric Acid Pump and OPEN CS-V-426, Emergency Borate Valve.

Proposed Answer: A

A is correct. The response not obtained action for the first step in FR-S.1 (immediate action step) directs a manual trip of the reactor. If the reactor will not trip manually then the step directs the operator to verify that control rods are being inserted in auto OR manually insert control rods.

B is incorrect but plausible. Step 2 of the procedure directs closing the MSIV's if the turbine had not tripped. Additionally, step 15 of the procedure directs allowing the RCS to heat up in order to insert negative reactivity in the event that a boration source were not available. It is plausible that closing the MSIV's would insert negative reactivity as it would isolate the steam dumps, however this is not a specific strategy delineated in the procedure.

C is incorrect but plausible. Aligning the charging pump suction to the RWST and isolating the VCT suction source is plausible as it would introduce a more concentrated boration source into the RCS. This action is part of the FR-S.1 procedural strategy for inserting negative reactoivity, however it occurs after the immediate action steps of the procedure.

D is incorrect but plausible. Starting a boric acid pump and opening the emergency borate valve is a specific procedural strategy for inserting negative reactivity however the strategy occurs after the immediate action steps of the procedure.

Technical Reference(s):	FR-S.1, Response to Nuclear Power Generation/ATWS		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	029 ATWS
Question Source:	Modified-Braidwood 2007 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.7/45.5/45.6
Learning Objective:	L1200I01, L1200I02

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 7	Group #		1		
	K/A #		040 W/E12 Steam Line Rupture-Excessive Heat Transfer. AA1 Ability to operate and/or monitor the following as they apply to Steam Line Rupture: AA1.20 Containment Pressure and Temperature Trends.		
	Importance Rating		2.6		

Proposed Question:

Given the following plant conditions:

- Reactor Trip has occurred.
- Safety Injection has actuated.
- Main Steamline Isolation has actuated.
- RCS pressure is 1820 psig and decreasing rapidly.
- RCS Tavg is 530°F and decreasing rapidly.
- Containment humidity is increasing.
- Main Steamline, SG Blowdown and Condenser Off-Gas Radiation Monitors all read normal.
- Containment pressure is 2.4 psig and increasing.
- Containment radiation is normal.

What event occurred?

A. A LOCA inside containment.
B. A LOCA outside containment.
C. A faulted steamline inside containment.
D. A faulted steamline outside containment.

Proposed Answer: C

C is correct. The given conditions are indicative of a faulted steam line inside containment. Safety Injection would have actuated based on low steam line pressure. Additionally, the fault would have resulted in decreasing RCS pressure and an RCS cool down due to steam flow through the fault. A fault in containment would cause an increase in containment pressure and humidity. A steam line fault in containment would not result in an increase in containment radiation levels. Additionally, a steam line fault would not affect Main Steam line, SG Blowdown and Condenser Offgas Radiation Monitors.

A is incorrect but plausible. A LOCA inside containment would cause a Safety Injection signal and could result in an RCS temperature decrease as cool ECCS water is injected to the loops. Additionally containment pressure and humidity would increase however the containment radiation level would also increase.

B is incorrect but plausible. A LOCA outside of containment would cause a Safety Injection signal and could result in an RCS temperature decrease as cool ECCS water is injected to the loops however containment humidity and pressure would not be increasing.

D is incorrect but plausible. A faulted steam line outside of containment would cause a Safety Injection and would have resulted in decreasing RCS pressure and an RCS cool down however containment humidity and pressure would not be increasing.

Technical Reference(s):	E-0, Reactor Trip or Safety Injection, Steps 9-11.		Westinghouse-Precautions, Limitations and Setpoints for Nuclear Steam Supply Systems-Seabrook. Setpoints, Part B-I, Reactor Protection System, 1, Safeguards Actuation.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	040 W/E12 Steam Line Rupture-Excess Heat Transfer
Question Source:	Modified-Seabrook 2007 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.5/45.6
Learning Objective:	L1202I02, L1207I01

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 8	Group #		1		
	K/A #		054 Loss of Main Feedwater 2.4.21 Knowledge of parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactive release control, etc.		
	Importance Rating		4.0		

Proposed Question:

The crew is responding to a LOCA and has entered E-1, "Loss of Reactor or Secondary Coolant." The following plant conditions exist:

- RCS pressure is 470 psig and stable.
- Containment pressure is 15 psig and increasing.
- Total available EFW flow is 350 gallons per minute.
- No operator action has been taken.
- Steam Generator pressures are:
  - SG "A": 800 psig and slowly decreasing.
  - SG "B": 800 psig and slowly decreasing.
  - SG "C": 950 psig and stable.
  - SG "D": 950 psig and stable.
- Steam Generator Wide Range levels are:
  - SG "A": 55% and decreasing.
  - SG "B": 45% and decreasing.
  - SG "C": 59% and slowly increasing.
  - SG "D": 59% and slowly increasing.

What action is required?

A. Remain in E-1, "Loss of Reactor or Secondary Coolant", and verify Containment Building Spray actuation.

B. Remain in E-1, "Loss of Reactor or Secondary Coolant", and take actions to restore adequate

Steam Generator levels.

C. Enter FR-H.1, "Response to Loss of Secondary Heat Sink", and immediately apply steps to establish RCS Feed and Bleed.

D. Enter FR-H.1, "Response to Loss of Secondary Heat Sink"' and transition back to E-1, Loss of Reactor or Secondary Coolant after determining that Secondary Heat Sink is not required.

Proposed Answer:	D	

This question tests the students knowledge of the parameters and logic associated with the FR-H safety function as well as the specific condition (based on operator controlled feedwater flow) when utilization of the FR-H.1 safety function response procedure is not warranted.

D is correct. The safety function parameters used to assess FR-H include Steam Generator inventory and total feedwater flow to the Steam Generators. Containment pressure of 15 psig meets the criteria for adverse containment conditions (>4psig). For adverse containment conditions the heat sink criteria of at least one Steam Generator >15% Narrow Range is not met, as Wide Range levels less than 60% would be off scale low on Narrow Range. With Steam Generator inventory criteria not met Emergency Feedwater Flow Red Path criteria requires >500 gpm total flow. The total EFW flow criteria is not met, thus an FR-H Red Path exists and FR-H.1 must be applied.

Further FR-H safety function criteria are assessed at Step 1 of procedure FR-H.1 where the need for Secondary Heat Sink is assessed. The parameters used for determining the secondary heat sink requirement are RCS pressure and non-faulted Steam Generator Pressures. If RCS pressure is not greater than any non-faulted Steam Generator pressure then the action is to "Return to Procedure and Step in Effect". Given the parameters stated in the stem of the question the correct assessment is that Secondary Heat Sink is not required and the operators should return to procedure E-1.

A is incorrect but plausible. It is plausible that FR-H.1 entry conditions are not met because a heat sink is not required (reactor coolant pressure is less then steam generator pressure). The FR-H.1 entry conditions are not conditional based on the necessity of the secondary heat sink. FR-H.1 must be entered. The determination of secondary heat sink requirement is made after FR-H.1 is entered. Step 1 of FR-H.1 checks for secondary heat sink requirement. If secondary heat sink is not required then the procedure directs "Return to procedure and step in effect". At that point a transition would be made back to procedure E-1. There is a common operator misconception regarding the entry condition requirements of procedure FR-H.1 if a) reactor coolant pressure is less than steam generator pressure and b) feed flow to the steam generators is less than 500 gpm due to operator action.

B is incorrect but plausible. Total EFW flow is one of the parameters used to assess the requirement to execute FR-H.1. If total EFW flow is <500 gpm "due to operator action" then FR-H.1 is not entered. The stem of the question states that total available EFW flow is 350 gpm. This condition could be misinterpreted as being total flow due to operator action. In this case the candidate may decide that the correct action is to recover adequate SG levels per guidance of procedure E-1. There is a common operator misconception regarding the entry condition requirements of procedure FR-

H.1 if a) reactor coolant pressure is less than steam generator pressure and b) feed flow to the steam generators is less than 500 gpm due to operator action.

C is incorrect but plausible if the candidate determines that FR-H.1 should be performed and misapplies the Bleed and Feed criteria. The Bleed and Feed criteria is based on Steam Generator level of "any 3 SG levels less than 51% for adverse containment". The candidate could misapply the FR-H.1 safety function status criteria (65% Wide Range level) and errantly determine that bleed and feed is required.

Technical Reference(s):	FR-H.1, Response to Loss of Secondary Heat Sink		F-0.3, Heat Sink (H) Status Tree Logic

Proposed references to be provided to applicants during examination:	None
K/A Topic:	054 Loss of Main Feedwater
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/43.5/45.12
Learning Objective:	L1211I01

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 9	Group #		1		
	K/A #		055 Station Blackout EK3 Knowledge of the reasons for the following responses as they apply to the Station Blackout. EK3.02 Actions contained in EOP for loss of offsite and onsite power.		
	Importance Rating		3.3		

Proposed Question:

What is the basis for the strategy of depressurizing the intact Steam Generators at maximum allowable rate in procedure ECA-0.0, "Loss of All AC Power?"

A. To prevent saturation of reactor coolant by maintain subcooling greater than 40°F.
B. To minimize reactor coolant inventory loss through the reactor coolant pump seals.
C. To enhance emergency feedwater flow capability from the steam driven EFW pump.
D. To minimize core cooling challenges while forced reactor coolant flow is unavailable.

Proposed Answer: B

B is correct. Step 17 of ECA-0.0 prescribes depressurizing the intact Steam Generators to reduce RCS leakage. Per the Westinghouse ECA-0.0 background document the intact steam generators are depressurized "thereby reducing RCS temperature and pressure to reduce RCP seal leakage and minimize RCS inventory loss.

A is incorrect but plausible. The steam generators are depressurized in part to reduce RCS temperature, however this is pursuant to minimizing RCP seal degradation due to elevated temperature. There is no guideline in step 17 pursuant to maintaining 40°F subcooling. A note prior to step 17 states that voiding in the vessel upper head region may occur and that this should not prevent continued depressurization.

C is incorrect but plausible. Depressurizing the steam generators would reduce backpressure on the EFW lines and allow for increased EFW flow capability, however this is not included in the basis for the depressurization.

D is incorrect but plausible. The steam generators are depressurized in part to reduce RCS temperature however this is pursuant to minimizing RCP seal degradation due to elevated temperature. It is plausible that core cooling would be a concern when forced flow cooling is not available and the plant must rely on natural circulation. If this were true then depressurizing the steam generators would be a plausible strategy to enhance natural circulation.

Technical Reference(s):	ECA-0.0, Loss of All AC Power, Step 17		Westinghouse Background Document, ECA-0.0, pgs. 7-8, 71, and 120-125.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	055 Station Blackout
Question Source:	Bank. Seabrook 2007 Company Audit Exam. Originally modified from Byron 1998 NRC Exam.
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.13
Learning Objective:	L8067I03, L8067I04, L8067I10

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 10	Group #		1		
	K/A #		056 Loss of Offsite Power AA1 Ability to operate and/or monitor the following as they apply to the Loss of Offsite Power: AA1.21 Reset of ESF Load Sequencers.		
	Importance Rating		3.3		

Proposed Question:

Given the following conditions:

- A loss of offsite power occurred.
- BUS E6 was re-energized by its Emergency Diesel Generator.
- The BUS E5 Emergency Power Sequencer activated however Bus E5 was not re-energized due to an internal fault in the sequencer.
- The crew has verified that offsite power has been restored and is stable.

What action must the control board operator take to restore power to Bus E5 and deactivate its Emergency Power Sequencer?

A. Reset RMO and select the CLOSE position on either the BUS E5 UAT or RAT breaker control switch.

B. No operator action is necessary. Once offsite power was restored the BUS E5 RAT breaker automatically closed.

C. Hold the RMO Bypass switch in the BYPASS position and select the CLOSE position on either the BUS E5 UAT or RAT breaker control switch.

D. Select either the UAT or RAT position on the BUS E5 synchronizing selector switch and select the CLOSE position on the associated breaker control switch.

Proposed Answer: C

C is correct. The RMO Bypass Switch is designed to allow for bypassing the associated bus UAT and RAT breaker trip and lockout contacts in the event that the Emergency Power Sequencer activates but does not complete sequencing, the associated emergency diesel generator does not start or the associated diesel generator breaker does not close and energize the bus. The Emergency Power Sequencer is deactivated and reset by re-energizing the associated bus from the UAT or RAT.

A is incorrect but plausible. It is true that re-energizing the bus from either the UAT or RAT will reset the Emergency Power Sequencer however resetting RMO will not allow for UAT or RAT breaker closure. RMO cannot be reset until step 9 of the sequencing process is complete. The stem of the question stated that the associated emergency diesel did not start thus the sequencing process would not progress to step 9. In this case the RMO interlock would still be active and the UAT and RAT breakers would have a trip and lockout signal preventing their closure. B is incorrect but plausible. The RAT breaker is designed to automatically close if the associated bus is powered from the UAT and the UAT trips open, provided there is offsite power available to the RAT. The conditions in the stem of the question state that offsite power was lost and that the Emergency Power Sequencer was activated. The RAT breaker would have an RMO trip and lockout signal preventing its closure. D is incorrect but plausible. With offsite power available it is conceivable that the operator would have to utilize the synch check circuit however the associated bus is de-energized so no synch check is needed. Additionally, to close the UAT or RAT breaker in this condition the RMO Bypass Switch would have to be utilized.

Technical Reference(s):	1-NHY-310102, Sheet A51b, 4160V Bus 1-E5 UAT Inc Line Close Schematic.		1-NHY-310102, Sheet A51c, Bus 1-E5 UAT Inc Line Trip Schematic.

Proposed references to be provided to applicants during examination:	None
K/A Topic:	056 Loss of Offsite Power
Question Source:	Modified from bank.
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	43.5/45.13
Learning Objective:	L8020I07, L8020I08, L8020I10, L8020I11, L8020I15

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 11	Group #		1		
	K/A #		057 Loss of Vital AC Instrument Bus AA2 Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Power. AA2.15 That a loss of ac has occurred.		
	Importance Rating		3.8		

Proposed Question:

The following events have occurred:
- Auto rod withdrawal and turbine loading is blocked.
- 2 of 4 feedwater regulating valves have positioned to full open.
- The 'B' PORV has armed but is not open.
- Control rods are continuously inserting.

Which of the following vital instrument panels has lost power?

A. Vital Instrument Power Panel PP-1A

B. Vital Instrument Power Panel PP-1D

C. Vital Instrument Power Panel PP-1E

D. Vital Instrument Power Panel PP-1F

Proposed Answer: A

A is correct. Per Dwg 310105, System Failure Analysis, EDE-PP-1A, a loss of PP-1A will cause those feedwater regulating valves selected for "Channel 1" level input to fail open, will block auto rod withdrawal and turbine loading, and will arm the B PORV. Additionally, a loss of PP-1A will result in FW-PT-505 failing low which will cause control rods to continuously insert.

B is incorrect but plausible. Loss of PP-1D can inhibit auto operation of both PORV's, but will not arm the B PORV nor will it cause rod insertion.

C is incorrect but plausible. Loss of PP-1E will result in loss of charging and seal injection flow control, loss of letdown. The loss of PP-1E will result in loss of a multitude of primary system functions. Loss of PP-1E has no impact on the rod control system.

D is incorrect but plausible. Loss of PP-1F will result in a multitude of conditions such as loss of PCCW temp. control, loss of letdown etc. but will not arm the B PORV, cause feedwater regulating valves to fail open or impact the rod control system.

Technical Reference(s):	OS1247.01, LOSS OF 120VAC INSTRUMENT PANEL 1A, 1B, 1C, OR 1D.		OS1247.02, LOSS OF 120VAC INSTRUMENT BUS PP-1E OR PP-1F.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	057 Loss of Vital AC Instrument Power

Question Source:	Bank. Seabrook 2007 NRC Exam			

Question Cognitive Level:	Higher: Comprehension/Analysis		

10 CFR Part 55 Content:	43.5/45.13	
Learning Objective:	L1186I09, L1186I12	

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 12	Group #		1		
	K/A #		058 Loss of DC Power AK3 Knowledge of the reasons for the following responses as they apply to the Loss of DC Power: AK3.02 Actions contained in EOP for Loss of DC Power.		
	Importance Rating		4.0		

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- All plant control systems are aligned normally.
- An electrical fault has occurred and VAS alarm D6096, "DC Bus 11B Volt Lo-Lo" has actuated.
- Train 'B' PCCW system temperature is slowly decreasing.
- Tavg/Tref is at +0.25°F and stable.
- The crew has entered procedure OS1248.01, "Loss of a Vital 125 VDC Bus."

What initial action does OS1248.01 direct?

A. Check Condenser Steam Dump status. If the Steam Dumps fail open then close the dumps by placing either Steam Dump Interlock Control Switch to the OFF position.

B. Check Rod Control status and take manual control if necessary. If Control Rods continue to insert then trip the reactor and go to E-0, "Reactor Trip or Safety Injection".

C. Check Feedwater control status and take manual control if necessary. If Feedwater control is not available then trip the reactor and go to E-0, "Reactor Trip or Safety Injection".

D. Check the status of the PCCW Containment Isolation Valves. If PCCW to containment is lost then trip the reactor and go to E-0, "Reactor Trip or Safety Injection" and trip the affected Reactor Coolant Pumps within 10 minutes.

Proposed Answer: C

C is correct. OS1248.01 contains a caution statement that specifically states that a loss of DC Bus 11A or 11B will result in a loss of normal feedwater control. The first step in the procedure specifically directs monitoring of feedwater control, taking manual control if necessary and tripping the reactor if feedwater control is not available.

A is incorrect but plausible. Condenser steam dumps are affected by various electrical system failures however they pertain to Non-Vital 125 VDC and Vital 120VAC. Procedure ON1248.02, Loss of Non Vital 125 VDC contains a step for taking manual control of condenser steam dumps however it is not an initial major action. Additionally, procedure OS1247.01, Loss of a 120 VAC Instrument Panel contains procedural guidance for closing the steam dumps using either Bypass Interlock switch if the steam dumps are failed open.

B is incorrect but plausible. A loss of vital AC would require placing Control Rods in manual. This strategy is directed by step 1 of OS1247.01, “Loss of a 120 VAC Instrument Panel”. There is no such action required for a loss of DC event.

D is incorrect but plausible. This distracter is plausible as Control Room Operators are aware of the sensitivity of RCP cooling in the event that PCCW cooling is lost to the pumps. OS1248.01 does address the affect of a loss of DC on the PCCW system. The affect is a loss of temperature control vice an isolation of PCCW to containment. There is a common operator misconception regarding the failure mode of the PCCW temperature control valves. The temperature control valves fail to the “full cooling” mode so there would not be a loss of cooling concern for the RCP’s.

There is a generic AOP/EOP action for a loss of PCCW cooling to the RCP’s that requires the pumps to be removed from service within 10 minutes. This action is not included in OS1248.01. The PCCW containment isolation valves have DC “latching” relays such that the valves fail to the last position on a loss of DC power. In this case the containment isolation valves would fail in the open position which would not result in a loss of cooling to the RCP’s.

Technical Reference(s):	OS1248.01, Loss of a Vital 125 VDC Bus		ON1248.02, Loss of Non Vital 125VDC Bus
	OS1247.01, Loss of a 120 VAC Instrument Panel		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	058 Loss of DC Power
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.1
Learning Objective:	L1189I02, L1189I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 13	Group #		1		
	K/A #		062 Loss of Nuclear Service Water AA1 Ability to operate and/or monitor the following as they apply to the loss of Nuclear Service Water: AA1.02 Loads on the SWS in the control room.		
	Importance Rating		3.2		

Proposed Question:

Given the following plant conditions:

- The plant is initially at 100% power.
- A Tower Actuation (TA) signal is received on BOTH Service Water trains.
- The crew enters OS1216.01, "Degraded Ultimate Heat Sink" and is verifying proper alignment for Tower Actuation.
- All associated equipment realigned as expected.

What Service Water loads have isolated?

A. Emergency Diesel Generator heat exchangers.

B. PCCW heat exchangers.

C. SCCW heat exchangers.

D. PAB Fire Protection Booster Pump (FP-P-374).

Proposed Answer: C

C is correct. Since BOTH trains of Service Water have received a TA signal then the turbine building train related SW isolation valves (SW-V-4 and SW-V-5) will have closed. This will isolate SW to the SCCW heat exchanger.

A is incorrect but plausible. The Emergency Diesel Generator heat exchanger does have automatic isolation valves however they are designed to open upon a start of the EDG.

B is incorrect but plausible. The PCCW heat exchangers do have automatic isolation valves however they are designed to open and prevent manual closure upon a TA signal.

D is incorrect but plausible. The PAB Fire Protection Booster Pump (FP-P-374) supply is from the SW system within the PAB. It is plausible that the FP booster pump subsystem would be isolated in the event of a TA to prevent potentially pumping down the cooling tower inventory. There is no automatic isolation of this subsystem.			
Technical Reference(s):	1-NHY-503956, SW to DG WTR Jacket HX Logic Diagram		1-NHY-503977, SW Isol Valves for SW Flow to Turbine Bldg
	1-NHY-503974, SW-CCW-HX Discharge Valves Logic Diagram		PID-1-SW-B20795, Service Water System Nuclear Detail
Proposed references to be provided to applicants during examination:			None
K/A Topic:	062 Loss of Nuclear Service Water		
Question Source:	Bank. Seabrook 2005 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7/45.5/45.6		
Learning Objective:	L8037I13		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 14	Group #		1		
	K/A #		065 Loss of Instrument Air 2.1.32 Ability to explain and apply system limits and precautions.		
	Importance Rating		3.2		

Proposed Question:

Given the following plant conditions:

- A plant startup is in progress following a refueling outage.
- The plant is at 10% power and increasing at 3% per hour.
- Instrument air pressure is at 92 psig and decreasing.
- The crew has entered ON1242.01, "Loss of Instrument Air."

ON1242.01 contains criteria to trip the reactor if a loss of plant control occurs due to a loss of instrument air. For the given plant conditions which of the following describes a component failure that would lead to a loss of plant control, and why?

A. The main feedwater regulating valves fail open. This condition would result in a feedwater isolation due to high steam generator levels.

B. CS-HCV-182, Seal Injection Flow Control Valve fails open. This condition would result in an increase in seal injection flow to the reactor coolant pumps.

C. The main feedwater regulating bypass valves fail closed. This condition would result in a degradation of feedwater flow to the steam generators.

D. The PCCW heat exchanger outlet and bypass valves fail to the full bypass position. This condition would result in overheating the PCCW system.

Proposed Answer: C

C is correct. At 10% power the feedwater regulating bypass valves would still be aligned as a feedwater flowpath to the steam generators. These valves fail closed on a loss of instrument air, resulting in a degradation of feedwater flow to the steam generators.

A is incorrect but plausible. At 10% power the main feedwater regulating block valves would be open in preparation for transferring feedwater control from the bypass regulating valves to the main regulating valves. In this condition the main feedwater regulating valves are aligned as a potential flowpath to the steam generators. This answer is incorrect because the main feedwater regulating valves fail closed on a loss of instrument air.

B is incorrect but plausible. It is true that a loss of instrument air would cause CS-HCV-182 to fail open. The valve diverts flow to the seals via backpressure. If the valve were failed open there would be a decrease in seal injection flow to the reactor coolant pumps.

D is incorrect but plausible. The PCCW temperature control valves do assume a failed position, however they would align to the full cooling position. In this case the concern would be related to overcooling the PCCW system vice overheating.

Technical Reference(s):	ON1242.01, Loss of Instrument Air		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	065 Loss of Instrument Air

Question Source:	New			
Question Cognitive Level:	Higher: Comprehension/Analysis			
10 CFR Part 55 Content:	41.10/43.2/45.12			
Learning Objective:	L1194I03, L1194I04			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 15	Group #		1		
	K/A #		W/E04 LOCA Outside Containment EK1 Knowledge of the operational implications of the following concepts as they apply to the LOCA outside containment: EK1.3 Annunciators and conditions, indicating signals, and remedial actions associated with the LOCA outside containment.		
	Importance Rating		3.5		

Proposed Question:

Given the following:

- Reactor trip with Safety Injection from 100% power.
- The crew has entered E-0, "Reactor Trip or Safety Injection."
- Containment conditions:
  - ➢ Containment Radiation Monitors indicate normal and stable.
  - ➢ Containment pressure is normal and stable.
  - ➢ Containment Building Sump B level is 3.16 inches and stable.
- RHR Vault 'A' Area Radiation Monitor RM-6538-1 is reading 1.29 E+02 mR/Hr and increasing.
- PAB -6 Foot Elevation North Area Radiation Monitor RM-6508-1 is reading 4.49 mR/Hr and stable.
- RCS conditions:
  - ➢ RCS pressure is 1750 psig and decreasing.
  - ➢ Pressurizer level is 5% and decreasing.
  - ➢ PORV's are closed.
  - ➢ Pressurizer spray valves are closed.
- Steam Generator conditions:
  - ➢ All Steam Generator pressures are approximately 950 psig and slowly decreasing.
  - ➢ Steam Generator narrow range levels all indicate off scale low.
  - ➢ Secondary radiation is normal and stable.

What procedure should be entered?

A. E-1, "Loss of Reactor or Secondary Coolant"
B. ES-1.1, "SI Termination"
C. ECA-1.2, "LOCA Outside Containment"
D. ES-1.2, "Post LOCA Cooldown and Depressurization"

Proposed Answer:	C	

C is correct. Per the conditions in the question stem there is a loss of reactor coolant that is not inside containment and not to the secondary side. The determination of a LOCA outside containment is based on radiation indications in auxiliary buildings.

A is incorrect but plausible. The conditions in the stem of the question are indicative of a loss of reactor coolant however the specific indications utilized as entry conditions for procedure E-1 are based on containment building radiation, pressure and level which are all normal.

B is incorrect but plausible. SI Termination is one of the major EOP transition points from E-0 however the ECCS termination criteria are not met as RCS pressure and level are decreasing.

D is incorrect but plausible. ES-1.2, Post LOCA Cooldown and Depressurization is one of the major procedural transitions from E-0 however the transition is based on stable pressurizer level.

Technical Reference (s):	E-0, "Reactor Trip or Safety Injection".		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	W/E04 LOCA Outside Containment
Question Source:	Bank. Edited from DC Cook 2001 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.8/41.10/45.3
Learning Objective:	L1209I04

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 16	Group #		1		
	K/A #		038 Steam Generator Tube Rupture (SGTR) EK3 Knowledge of the reasons for the following responses as they apply to the SGTR: EK3.06 Actions contained in EOP for RCS water inventory balance, S/G tube rupture, and plant shutdown procedures.		
	Importance Rating		4.2		

Proposed Question:

Given the following plant conditions:

- A steam generator tube rupture has occurred concurrent with a loss of off-site power.
- All safeguards systems functioned as designed.
- Actions of E-3, "Steam Generator Tube Rupture" have been performed.
- The crew is preparing to cool down and depressurize the RCS to Mode 5.

What cooldown method is preferred and why?

A. ES-3.1, "Post SGTR Cooldown Using Backfill." Minimizes radiological release.
B. ES-3.3, "Post SGTR Cooldown Using Steam Dump." Minimizes radiological release.
C. ES-3.3, "Post SGTR Cooldown Using Steam Dump." Provides fastest cooldown.
D. ES-3.1, "Post SGTR Cooldown Using Backfill." Provides fastest cooldown.

Proposed Answer: A

A is correct. Backfilling from the steam generators is the preferred available method for cooldown. Per Westinghouse Background Document, E-3 "Post-SGTR cooldown using backfill is the preferred method since it minimizes radiological releases and facilitates processing of contaminated primary coolant". The background document further states that this method is slow as compared to Post SGTR Cooldown Using Steam Dump. The benefit of utilizing the backfill method to minimize radiological releases is enhanced by the fact that power is not available to support condenser steam dump operation per the conditions in the question stem.

B is incorrect but plausible. Cooling down with steam dumps is advantageous particularly if feedwater inventory (CST) becomes limited. Use of this method is not preferred as the condenser steam dumps are not available due to the loss of power. In this case the atmospheric dumps would have to be used and a radiological release would not be minimized.

C is incorrect but plausible. Cooling down by dumping steam is the fasted method however use of this method is not preferred as the condenser steam dumps are not available due to the loss of power. In this case the atmospheric dumps would have to be used and a radiological release would occur.

D is incorrect but plausible. Backfilling from the steam generators is the preferred available method for cooldown however it is because the method minimizes radiological releases. This method does not provide the fastest cooldown. The Westinghouse background document states that this method is slow as compared to using steam dumps.

Technical Reference(s):	E-3, "Steam Generator Tube Rupture"		Westinghouse Background Document, E-3, pg.169.
Westinghouse Background Document, ES-3.1, pg.12	Westinghouse Background Document, ES-3.2, pg.12		Westinghouse Background Document, ES-3.3, pg.12

Proposed references to be provided to applicants during examination:	None

K/A Topic:	038 Steam Generator Tube Rupture (SGTR)

Question Source:	Bank. TEB 2206		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.13		
Learning Objective:	L1205I03		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 17	Group #		1		
	K/A #		W/E05 Inadequate Heat Transfer-Loss of Secondary Heat Sink EK2 Knowledge of the interrelationships between the Loss of Secondary Heat Sink and the following: EK2.2 Facilities 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.		
	Importance Rating		3.9		

Proposed Question:

What is the consequence of having only one PORV open during implementation of the bleed and feed steps in FR-H.1 "Response to Loss of Secondary Heat Sink?"

A. Reactor coolant system pressure will continue to rise to the pressurizer safety valve setpoint leading to further loss of coolant inventory.
B. Insufficient bleed flow will inhibit mixing of Safety Injection flow leading to localized pressurized thermal shock conditions.
C. The reactor coolant system will not depressurize enough to allow for adequate reflux cooling between the loop hot legs and the steam generators.
D. The reactor coolant system will not depressurize enough to allow for adequate feed of subcooled SI flow to adequately remove core decay heat.

Proposed Answer: D

D is correct. Per the Westinghouse Background Document for FR-H.1 "If both PRZR PORV's are not maintained open, the RCS may not depressurize sufficiently to permit adequate feed of subcooled SI flow to remove core decay heat. If core decay heat exceeds RCS bleed and feed heat removal capability the RCS will repressurize rapidly, further reducing the feed of subcooled SI flow and resulting in a rapid decrease in RCS inventory.

A is incorrect but plausible. Having only one PORV open would reduce the ability to depressurize the RCS to ensure adequate SI flow however if one PORV is opened RCS pressure would decrease and not "continue to rise".

B is incorrect but plausible. Insufficient bleed flow is a valid concern however localized thermal shock is not a concern addressed by the FR-H.1 background document.

C is incorrect but plausible. FR-H.1 includes the strategy of stopping all reactor coolant pumps to remove pump heat input into the RCS so reflux cooling may be considered a possible condition. Reflux cooling is not a condition addressed by the FR-H.1 background document.

Technical Reference(s):	FR-H.1, Response to Loss of Secondary Heat Sink.		Westinghouse Background Document, FR-H.1, pgs 10 and 99.

Proposed references to be provided to applicants during examination:	None
K/A Topic:	W/E05 Inadequate Heat Transfer-Loss of Secondary Heat Sink
Question Source:	Modified from bank. 2006 Braidwood NRC Exam.
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.7/45.7
Learning Objective:	L1211I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 18	Group #		1		
	K/A #		077 Generator Voltage and Electrical Grid Disturbances. AK1 Knowledge of the operational implications of the following concepts as they apply to Generator Voltage and Electrical Grid Disturbances: AK1.03 Under-excitation.		
	Importance Rating		3.3		

Proposed Question:

Given the following plant conditions:

- An electrical grid disturbance occurs resulting in a loss of the 345kv Tewksbury (394) line.
- Generator reactive load is at 75 MVAR "Leading".
- Alarm point D6442 GEN UEL LIMIT-LIMITER ON is in alarm.

What action should be taken in accordance with the alarm response procedure?

A. Main generator excitation should be raised in order to increase VAR loading in the leading direction.

B. Main generator excitation should be raised in order to increase VAR loading in the lagging direction.

C. Main generator excitation should be lowered in order to increase VAR loading in the leading direction.

D. Main generator excitation should be lowered in order to increase VAR loading in the lagging direction.

Proposed Answer: B

B is correct. The conditions stated in the question stem are indicative of an under-excited condition in the main generator. As generator excitation is raised VAR loading will move in the "lagging" direction.

A is incorrect but plausible. Raising excitation is done such that VAR loading trends in the "lagging" direction.

C is incorrect but plausible. It is true that VAR loading must be adjusted however VARS should be adjusted in the “lagging” direction vice increasing “leading” VAR’s.

D is incorrect but plausible. It is true that VAR loading must be in the “lagging” direction however this would be accomplished by increasing generator excitation.

Technical Reference(s):	VAS Alarm Procedure D6442, GEN UEL LIMIT-LIMITER ON		MPCS, Generator Capability Curve

Proposed references to be provided to applicants during examination:	None

K/A Topic:	077 Generator Voltage and Electrical Grid Disturbances.

Question Source:	New			
Question Cognitive Level:	Higher: Comprehension/Analysis			
10 CFR Part 55 Content:	41.4/41.5/41.7/41.10/45.8			
Learning Objective:	L8016I03			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 19	Group #		2		
	K/A #		005 Inoperable/Stuck Control Rod AK1 Knowledge of the operational implications of the following concepts as they apply to the Inoperable/Stuck Control Rod: AK1.02 Flux Tilt		
	Importance Rating		3.1		

Proposed Question:

Given the following plant conditions:

- The plant was initially at 90% power.
- A power decrease to 70% power is in progress.
- During the power decrease Control Bank D rods were manually inserted from 220 steps to 200 steps.
- Subsequently the crew identifies that control rod H8 in Control Bank D did not move and is stuck at 220 steps.

Assuming NO operator action, if left uncorrected, what is the affect of misaligned control rod H8 on core neutron flux distribution?

A. Neutron flux will peak locally in the area of rod H8. This will cause hot channel concerns in all areas of the core.

B. The affect on the overall neutron flux will be minimal since flux at the tip of rod H8 is much smaller than average core flux.

C. Neutron flux will peak locally in the area of rod H8. This will only cause hot channel concerns in the local area around the stuck rod.

D. Neutron flux will be suppressed locally in the area of rod H8. This will cause an increase in neutron flux in all other areas of the core.

Proposed Answer: C

C is correct. A stuck rod that is misaligned high will cause localized neutron flux peaking in the area of the stuck rod. There would be a hot channel concern due to the localized flux peak and higher power production.

A is incorrect but plausible. It is true that flux will peak locally at the stuck rod and that hot channel concerns will exist local to the stuck rod. It is conceivable that the stuck rod could be seen as causing an overall positive reactivity affect with overall core flux increasing with resulting overall hot channel concerns.

B is incorrect but plausible. The flux at the tip of a dropped rod is small as compared to average core flux. This distractor could be chosen if the concept of flux affects of a dropped rod were applied.

D is incorrect but plausible. This distractor would be correct if the stuck rod were lower into the core than the rest of bank D.

Technical Reference(s):	OS1210.06, "Misaligned Control Rod", caution statement prior to step 2.		Tech Spec. 3.1.3.1 basis.

Proposed references to be provided to applicants during examination:	None
K/A Topic:	005 Inoperable/Stuck Control Rod
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.8/41.10/45.3
Learning Objective:	L8125I06

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 20	Group #		2		
	K/A #		028 Pressurizer Level Malfunction AA1 Ability to operate and/or monitor the following as they apply to the Pressurizer Level Control Malfunctions: AA1.07 Charging pump maintenance of PZR level (including manual backup)		
	Importance Rating		3.3		

Proposed Question:	

Given the following plant conditions:

- The plant is at 50% power.
- All control systems are in normal alignment.
- Pressurizer Level Control is selected to channels 459/461.

What Pressurizer level channel failure would initially cause an increase in charging flow?

A. LT-459 fails low
B. LT-461 fails low
C. LT-459 fails high
D. LT-461 fails high

Proposed Answer:	A	

A is correct. With the Pressurizer level channels selected to the 459/461 combination LT-459 is the controlling channel and feeds into the charging flow control valve CS-FK-121 which throttles flow from the discharge of the running centrifugal charging pump. If LT-459 fails low then a signal is sent to CS-FK-121 to open the valve and increase charging flow.

B is incorrect but plausible. It is a common operator misconception that failure low of the controlling or backup channel will cause charging flow to increase. There is a common failure affect from either the controlling or backup channel failure, however it is that the pressurizer heaters will de-energize and letdown flow will isolate. Only the controlling channel feeds into the charging flow control valve CS-FK-121 to control charging flow.

C is incorrect but plausible. A high failure of controlling channel LT-459 would initially cause charging flow to decrease. In this case pressurizer level would decrease and eventually cause a letdown isolation when pressurizer level drops to 17%. After the letdown isolation charging flow would then cause a level increase, however this would be a delayed affect.

D is incorrect but plausible. A high failure of backup channel LT-461 would not initially cause charging flow to decrease. In this case pressurizer level would remain on program and a high level alarm would be received. As stated in the plausibility statement for answer B, it is a common operator misconception that the primary and backup level channels have a common failure affect on charging flow control. If this misconception is applied then it is plausible that a high failure of LT-461 would cause a similar transient to the one described in the plausibility statement for answer C.

Technical Reference(s):	1-NHY-509027, Pressurizer Level Control Functional Diagram		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	028 Pressurizer Level Malfunction
Question Source:	Bank. 2007 Diablo Canyon NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.5/45.6
Learning Objective:	L8027I05, L8027I06, L8027I14

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 21	Group #		2		
	K/A #		032 Loss of Source Range NI's AA2 Ability to determine and interpret the following as they apply to Loss of Source Range Instrumentation: AA2.04 Satisfactory source-range/ intermediate-range overlap.		
	Importance Rating		3.1		

Proposed Question:

Given the following plant conditions:

- A plant shutdown is in progress.
- Reactor power is 6% and decreasing.
- Nuclear Instrumentation Intermediate Range Channel N-36 fails HIGH.

How will this failure affect the plant shutdown and subsequent operation of the Nuclear Instrumentation Source Range Channels?

A. The reactor will trip. The Source Range NI's will have to be manually energized.

B. The reactor will NOT trip. The Source Range NI's will have to be manually energized.

C. The reactor will NOT trip. The Source Range NI's will automatically energize when Intermediate Range Channel N-35 decreases to the proper setpoint.

D. The reactor will trip. The Source Range NI's will automatically energize when Intermediate Range Channel N-35 decreases to the proper setpoint.

Proposed Answer: A

A is correct. The reactor will trip when Intermediate Range Channel N-36 fails HIGH. An Intermediate Range reactor trip signal occurs when 1 of 2 Intermediate Range Channels is >25% equivalent current. A failed high IR channel would exceed the 25% equivalent current. This trip is active when below the P-10 setpoint (10% reactor power). The Source Range NI's will not automatically energize. During a reactor shutdown the Intermediate Range Channels will decrease down to the P-6 reset value of 5×10^{-11} amps at which time the Source Range Channels will energize within the normal 1 decade of overlap indication with the intermediate range. Automatic energization of the Source Range NI's requires <u>both</u> Intermediate Range NI's to reduce to the P-6 reset value.

B is incorrect but plausible. There are common operator misconceptions with regard to NI related permissive signals, particularly P-6, P-8, and P-10. The stem of the question states that reactor power is at 6% and decreasing. The candidate may incorrectly interpret this as being above P-8 (which is actually a permissive for a reactor trip based on a single loop loss of coolant flow with a setpoint of 50% on the Power Range NI instrumentation). If the candidate makes this error then they may interpret that the Intermediate Range channels are still above the interlock setpoint where they would initiate a reactor trip. This interlock is P-10 (setpoint 10%) as described above for answer A. The second half of the answer is correct.

C is incorrect but plausible. As stated in the plausibility statement for answer B, there are common operator misconceptions with regard to NI related permissive signals, particularly P-6, P-8, and P-10. The stem of the question states that reactor power is at 6% and decreasing. The candidate may incorrectly interpret this as being above P-8 (which is actually a permissive for a reactor trip based on a single loop loss of coolant flow with a setpoint of 50% on the Power Range NI instrumentation). If the candidate makes this error then they may interpret that the Intermediate Range channels are still above the interlock setpoint where they would initiate a reactor trip. This interlock is P-10 (setpoint 10%) as described above for answer A. Additionally, the candidate may have a misconception that the P-6 reset only requires 1 of 2 Intermediate Range channels to decrease to the P-6 reset value of $5x10^{-11}$amps, however this requires both channels to do so.

D is incorrect but plausible. The reactor will trip when Intermediate Range Channel N-36 fails HIGH. An Intermediate Range reactor trip signal occurs when 1 of 2 Intermediate Range Channels is >25% equivalent current. A failed high IR channel would exceed the 25% equivalent current. This trip is active when below the P-10 setpoint (10% reactor power). The candidate may have a misconception that the P-6 reset only requires 1 of 2 Intermediate Range channels to decrease to the P-6 reset value of $5x10^{-11}$amps, however this requires both channels to do so.

Technical Reference(s):	Precautions, Limitations and Setpoints for Nuclear Steam Supply Systems, pgs. 6 and 12.		E-0, Reactor Trip or Safety Injection, Attachment B, Reactor Trip Signals
	Seabrook MPC, Primary Tech Data, NI Comparison.		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	032 Loss of Source Range NI's		
Question Source:	Bank. Seabrook 2004 Company Exam.		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	43.5/45.13		
Learning Objective:	L8030I08, L8030I02, L8030I11		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 22	Group #		2		
	K/A #		037 Steam Generator Tube Leak AK3 Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: AK3.05 Actions contained in procedures for radiation monitoring, RCS water level inventory balance, S/G tube failure and plant shutdown.		
	Importance Rating		3.7		

Proposed Question:

Given the following plant conditions:

- A SG tube leak has occurred.
- The crew has entered OS1227.02, "Steam Generator Tube Leak."
- The crew has identified the affected SG.
- When isolating flow from the affected SG, its Main Steam Isolation Valve (MSIV) fails to close.
- Per procedure the crew closes the remaining MSIVs.

Why are the intact SG MSIVs closed?

A. Prevents contamination of the intact SGs.

B. Allows the affected SG to be depressurized with its ASDV.

C. Prevents the intact SG pressures from decreasing during the cooldown.

D. Establishes a pressure differential between the affected and intact SGs.

Proposed Answer: D

D is correct. The Steam Generator Tube Leak procedure contains a step for isolating flow from the affected steam generator. This action is taken to minimize radiological releases and to maintain pressure in the affected steam generator greater than pressure in the intact steam generators following subsequent cooldown of the reactor coolant system. Isolating the affected steam generator is necessary to establish a pressure differential between the affected generator and the intact generators in order to cool and then depressurize the reactor coolant system in order to stop primary to secondary leakage. The procedure step prescribes closing the intact MSIV's in the event that the affected steam generator MSIV will not close.

A is incorrect but plausible. The basis for isolating the affected steam generator is to minimize radiological releases from that generator. The basis for closing the unaffected steam generator MSIV's is not to minimize cross-contamination but to establish a pressure differential between the affected generator and the intact generators.

B is incorrect but plausible. Depressurizing steam generators is the procedurally prescribed method for cooling down the reactor coolant system however this is done by depressurizing the unaffected steam generators.

C is incorrect but plausible. Isolating the intact steam generators is to prevent depressurization of the affected steam generator vice the unaffected steam generators during the cooldown.

Technical Reference(s):	OS1227.02, Steam Generator Tube Leak, step 17		Westinghouse Background Document, ARG-3, Steam Generator Tube Leak, Basis for generic step 15, pgs 96-98.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	037 Steam Generator Tube Leak
Question Source:	Bank. Teb 31097
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.13
Learning Objective:	L1190I02

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 23	Group #		2		
	K/A #		051 Loss of Condenser Vacuum 2.4.11 Knowledge of abnormal condition procedures.		
	Importance Rating		4.0		

Proposed Question:

Given the following plant conditions:

- The crew entered procedure ON1233.01, "Loss of Condenser Vacuum" in response to decreasing condenser vacuum.
- Turbine generator load has been reduced to 360 MWE.
- Condenser Vacuum is 26 in. hg and stable.
- The source of the condenser vacuum leak has not been identified.

In accordance with ON1233.01, "Loss of Condenser Vacuum", which of the following actions should be taken by the crew?

A. Immediately trip the reactor and go to E-0, "Reactor Trip or Safety Injection."

B. Immediately trip the turbine and verify all stop valves close and the generator breaker opens.

C. Stop the turbine load decrease and continue with the procedure. If at any time condenser vacuum cannot be maintained greater than 25 in. hg then trip the reactor and go to E-0, "Reactor Trip of Safety Injection."

D. Continue the load reduction until the source of the leak is identified and the leak is isolated. If at any time condenser vacuum cannot be maintained greater than 25 in. hg then trip the reactor and go to E-0, "Reactor Trip of Safety Injection."

Proposed Answer: C

C is correct. Procedure ON1233.01, "Loss of Condenser Vacuum" step 3 prescribes reducing generator load until either a) load has been decreased to 360 MWe or condenser vacuum cannot be maintained greater than 25 in. hg. The conditions in the stem of the question meet these criteria. In this case the crew would continue in the procedure. The RNO for step 3 states that if condenser vacuum cannot be maintained greater than 25 in. hg and generator load has been reduced to 360 MWe then the reactor should be tripped and procedure E-0, "Reactor Trip or Safety Injection" should be entered.

A is incorrect but plausible. Procedure ON1233.01, Loss of Condenser Vacuum step 3 prescribes reducing generator load until either a) load has been decreased to 360 MWe or condenser vacuum cannot be maintained greater than 25 in. hg. It is plausible that the procedure would dictate immediately tripping the reactor once load had been reduced to 360 MWe, regardless of condenser vacuum, as operation of the turbine under low load is not desirable due to low pressure turbine blade heating concerns.

B is incorrect but plausible. The plant is below the P-9 Turbine Trip/Reactor Trip permissive so it is plausible that the turbine could be removed from service without tripping the reactor. The procedure prescribes tripping the reactor and going to E-0 as condenser vacuum is below the C-9 condenser availability permissive.

D is plausible if the candidate misapplies procedure step 3. Once generator load has been reduced to 360 MWe it should not be reduced any further.

Technical Reference(s):	ON1233.01, Loss of Condenser Vacuum		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	051 Loss of Condenser Vacuum
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.5/45.13
Learning Objective:	L1188I08

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 24	Group #		2		
	K/A #		074 Inadequate Core Cooling E06/EA2 Ability to determine and interpret the following as they apply to the Degraded Core Cooling: EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency conditions.		
	Importance Rating		3.4		

Proposed Question:

Which of the following lists plant conditions directly utilized by the Core Cooling Status Tree to determine the need to enter procedure FR-C.1, "Inadequate Core Cooling?"

A. Pressurizer level, RCP status, Reactor Vessel (RVLIS) level.
B. RCS Hot Leg temperature, RCS Cold Leg temperature, RCS pressure.
C. Core Exit Thermocouple temperature, Pressurizer level, RCP status.
D. RCS subcooling, Core Exit Thermocouple temperature, Reactor Vessel (RVLIS) level.

Proposed Answer: D

D is correct. The Core Cooling Critical Safety Function status tree utilizes RCS Subcooling, Core Exit Thermocouple temperature and RVLIS level to determine the need for implementation of FR-C.1 based on an Orange Path or Red Path condition.

A is incorrect but plausible. RVLIS level and RCP status are inputs into the Core Cooling status. It is plausible that Pressurizer level would be an input to the Core Cooling status as it would be an indicator that the reactor vessel is full however this parameter is utilized for the Inventory status.

B is incorrect but plausible. It is plausible that RCS Hot and Cold leg temperatures would input into the Core Cooling status as temperature is a direct indicator of heat removal from the core, however Core Exit Thermocouple temperatures are utilized instead as they are a more direct indicator of core cooling conditions. It is plausible that RCS pressure would be an input to Core Cooling status as that parameter is associated with subcooling; however the Core Cooling status utilizes the RCS Subcooling value derived from the RVLIS subsystem (Core Exit temp. vs. Wide Range RCS pressure).

C is incorrect but plausible. Core Exit temperatures and RCP status are utilized as input to the Core Cooling status. It is plausible that Pressurizer level would be an input to the Core Cooling status as it would be an indicator that the reactor vessel is full however this parameter is utilized for the Inventory status.			
Technical Reference(s):	F-0.2, Core Cooling (C)		F-0.3, Heat Sink (H)
	F-0.6, Inventory (I)		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	074 Inadequate Core Cooling		
Question Source:	New		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	43.5/45.13		
Learning Objective:	L1227I10		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 25	Group #		2		
	K/A #		W/E02 SI Termination EK2 Knowledge of the interrelations between the SI Termination and the following: EK2.2 Facilities heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems and the operation of the facility.		
	Importance Rating		3.5		

Proposed Question:

What criteria must be met in order to terminate Safety Injection following a small break LOCA?

A. Adequate RCS subcooling, Adequate Secondary Heat Sink, RCS Pressure stable or increasing, Adequate Pressurizer level.

B. Adequate RCS subcooling, RCS Pressure stable or increasing, Adequate Pressurizer level, RCS Hot Leg temperature stable or decreasing.

C. Adequate Secondary Heat Sink, Adequate RCS subcooling, RCS Pressure stable or increasing, RCS Hot Leg temperature stable or decreasing.

D. Adequate Secondary Heat Sink, RCS Pressure stable or increasing, Adequate Pressurizer level, Adequate Reactor Vessel Level.

Proposed Answer: A

A is correct. Per procedure E-1, "Loss of Reactor or Secondary Coolant" the following criteria must be met in order to transition to ES-1.1, SI Termination:

a. RCS Subcooling-Greater Than 40°F.
b. Secondary Heat Sink:
   - Total feed flow to intact SG's-Greater Than 500 GPM

     -or-

   - Wide range level in at least two intact SG's-Greater than 65% (narrow range level in at least one SG-Greater then 15% for Adverse Containment)

-or-

- Narrow range level in at least one intact SG- Greater than 6% (15% for Adverse Containment)

c. RCS pressure- Stable or Increasing
d. PZR level- Greater then 7% (28% for Adverse Containment)

B is incorrect but plausible. The thermal condition of the reactor coolant is a valid consideration. It is plausible that there would be a requirement for coolant temperature to be stable or decreasing prior to terminating Safety Injection however this is not a criterion. The thermal condition of the reactor coolant is addressed by the subcooling criteria.

C is incorrect but plausible. The thermal condition of the reactor coolant is a valid consideration. It is plausible that there would be a requirement for coolant temperature to be stable or decreasing prior to terminating Safety Injection however this is not a criterion. The thermal condition of the reactor coolant is addressed by the subcooling criteria.

D is incorrect but plausible. It is plausible that there would be a requirement for adequate reactor prior to terminating Safety Injection as this could be an indicator of any voiding in the vessel head, which could be the case with a steam space LOCA however this is not one of the criteria.

Technical Reference(s):	E-1, Loss of Reactor or Secondary Coolant, step 6		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	W/E02 SI Termination
Question Source:	Bank. Beaver Valley 1997 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.7/45.7
Learning Objective:	L1203I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 26	Group #		2		
	K/A #		W/E13 Steam Generator Overpressure EK3 Knowledge for the reasons for the following responses as they apply to the steam generator overpressure: EK3.3 Manipulations of controls required to obtain desired results during abnormal and emergency situations.		
	Importance Rating		3.2		

Proposed Question:

Given the following plant conditions:

- The crew has entered FR-H.2, "Response to Steam Generator Overpressure."
- Steam Generator pressures are:
  - Steam Generator 'A': 1100 psig.
  - Steam Generator 'B': 1240 psig.
  - Steam Generator 'C': 1100 psig.
  - Steam Generator 'D': 1100 psig.

Why does Step 5 of FR-H.2, "Response to Steam Generator Overpressure" direct reducing Steam Generator 'B' pressure to less than 1225 psig?

A. To reduce RCS temperature in order to ensure primary system integrity.

B. To prevent lifting the Steam Generator code safety valves and causing a radiological release.

C. To decrease pressure below the highest steamline safety valve setpoint to ensure secondary integrity.

D. To maintain Steam Generator pressure low enough to ensure adequate total emergency feedwater flow.

Proposed Answer: C

C is correct per the Westinghouse basis document for FR-H.2.

A is incorrect but plausible. Part of the overall strategy of FR-H.2 is to reduce RCS temperature however the purpose of that step is to mitigate any excessive heat transfer that may be causing the overpressure condition in the steam generator. The primary system integrity concern is also plausible as the FR-H critical safety function is associated with integrity of the primary plant side however this concern is associated with FR-H.1.

B is incorrect but plausible. Reducing steam generator pressure is associated with steam generator safety valve performance however it is to ensure that pressure is within the safety valve relieving capacity versus being below the lowest valve actuation point. The concern regarding radiological release is also plausible as the safety valves offer a direct release path, however this concern is associated with the condition where there is elevated radiation as in the case of a steam generator tube rupture.

D is incorrect but plausible. The FR-H safety function does take into account the need for adequate feed flow to the steam generators however the generator pressure concern addresses the need to isolate feed flow to the effected generator to remove it as a contributing factor to the over pressurization.

Technical Reference(s):	Westinghouse Background Document, FR-H.2.		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	W/E13 Steam Generator Overpressure
Question Source:	New
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.13
Learning Objective:	L1211I06

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		1		
Question 27	Group #		2		
	K/A #		W/E08 RCS Overcooling-PTS EK3 Knowledge of the reasons for the following responses as they apply to the Pressurized Thermal Shock: EK3.2 Normal, abnormal and emergency operating procedures associated with Pressurized Thermal Shock.		
	Importance Rating		3.6		

Proposed Question:

Why is it desirable to terminate SI in FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition" if the criteria are met?

A. To conserve water in the RWST.

B. SI flow may have contributed to the RCS cooldown.

C. RCS heat removal is via the steam generators and SI flow is not required.

D. The other SI termination criteria will have already been met when FR-P.1 is entered.

Proposed Answer: B

B is correct. Per the Westinghouse Background Document, SI flow is a significant contributor to any cold leg temperature decrease. It can also be a significant contributor to an overpressure condition if the RCS is intact. A check for SI termination is performed early in FR-P.1 based on less restrictive criteria than in other SI termination steps in the recovery guidelines to try to remove its unfavorable PTS effects.

A is incorrect but plausible. The Westinghouse background document Generic Issues: SI Reduction discusses SI termination/reduction during "smaller failure" scenarios. The document states "For smaller failures, RCS pressure may stabilize at a value which is significantly above conditions for operation of the Residual Heat Removal system. Reducing RCS pressure using only pressurizer pressure control would decrease leakage and increase SI flow. Eventually, the pressurizer would fill with water rendering pressurizer pressure control ineffective. Consequently***, in order to decrease RCS pressure to conserve makeup water supply***, establish RHR cooling and minimize any leakage outside containment, SI flow must be decreased.

FR-P.1 usage could apply to a situation where there is a “smaller failure” as described above where RCS pressure is at the described elevated condition. C is incorrect but plausible. It is true that there may be heat removal taking place via the steam generators, in fact Step 2 of FR-P.1 is designed to stop the RCS cooldown and includes actions for stopping heat removal via the ASDV’s or steam dumps. It is not necessarily true that SI is not required. A small break LOCA condition could exist where SI flow could not be terminated, in which case attempts are made to start an RCP. D is incorrect but plausible. The SI termination criteria in FR-P.1 is less restrictive than the criteria in the other EOP’s. It is not necessarily true that other EOP SI termination criteria would be met.

Technical Reference(s):	FR-P.1, Response to Imminent Pressurized Thermal Shock Condition.		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	W/E08 RCS Overcooling-PTS

Question Source:	Bank. Diablo Canyon 2009 NRC Exam			

Question Cognitive Level:	Memory or Fundamental Knowledge		

10 CFR Part 55 Content:	41.5/41.10/45.6/ 45.13	

Learning Objective:	L1208I05

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 28	Group #		1		
	K/A #		003 Reactor Coolant Pump A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RCP's and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.05 Effects of VCT pressure on RCP seal leakoff flows.		
	Importance Rating		2.5		

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- Subsequently the Reactor Operator performs a routine CVCS Volume Control Tank divert per OS1000.10, Figure 13:VCT divert to PDT's.
- During the divert evolution the Reactor Operator notices that the 'A' Reactor Coolant Pump #1 seal return flow is 2.5 gpm and slowly rising.

Why is the 'A' RCP #1 seal return flow rising and what action should the operator take?

A. VCT pressure has increased causing an increase in both seal injection and seal return flow. The operator should maintain VCT pressure less then 25 psig.
B. VCT pressure has decreased causing a decrease in #1 seal return backpressure. The operator should maintain VCT hydrogen pressure greater than 15 psig.
C. The VCT divert flowpath branches off of the charging flowpath causing a reduction in both seal injection and seal leakoff flow. The operator should adjust CS-LK-185, VCT Divert Control to maintain adequate seal injection flow.
D. The VCT divert flowpath branches off of the seal return line causing a decrease in #1 seal return backpressure. The operator should adjust CS-LK-185, VCT Divert Control to maintain adequate seal return backpressure.

Proposed Answer:	B	

B is correct. The Reactor Coolant Pump #1 Seal Return line is routed to the bottom or outlet of the VCT. VCT pressure has a direct impact on seal return backpressure. When the VCT is diverted the tank inlet flow from letdown is re-routed. This causes a resulting drop in VCT pressure. The drop in VCT pressure results in a drop in seal return backpressure and an increase in seal return flow. The procedural guidance for performing a VCT divert (procedure OS1000.10, Figure 13, VCT Divert to PDT's) directs the operator to verify that VCT pressure is being maintained greater than 15 psig.

A is incorrect but plausible. If VCT pressure increased there would be a resulting increase in charging pump suction head and a nominal increase in charging/seal injection flow. The divert evolution results in a decrease in VCT pressure vice an increase.

C is incorrect but plausible. If the divert flowpath were downstream of the charging pumps then there would be a resulting decrease in charging and seal injection flow with a nominal decrease in seal injection flow, however a divert flowpath at this location would cause a change in pressurizer level vice VCT level.

D is incorrect but plausible. If the divert flowpath did branch off of the seal return line then there would be a resulting decrease in seal return backpressure, however a divert flowpath at this location would cause a change in pressurize level vice VCT level.

This question relates to a plant specific training priority due to a human performance error event associated with CS-LK-185. Plant Action Request #00192766 documents a 2009 event when a board operator left CS-LK-185 in manual after performing a VCT divert evolution.

Technical Reference(s):	OS1000.10, Figure 13: VCT Divert to PDT's		PID-1-CS-B20725

Proposed references to be provided to applicants during examination:	None
K/A Topic:	003 Reactor Coolant Pump
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.13
Learning Objective:	L8024I08

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 29	Group #		1		
	K/A #		004 Chemical and Volume Control K1 Knowledge of the physical connections and/or cause effect relationships between the CVCS and the following systems: K1.26 Flowpaths from the CVCS to reactor coolant drain tank and holdup tank.		
	Importance Rating		2.7		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- All systems are functioning normally in their normal alignment.
- The letdown degassifier is not in service.
- Volume Control Tank level is at 81% and slowly increasing.

What is the status of letdown flow?

A. CS-LV-112A, Letdown Divert to BWST is THROTTLED OPEN diverting flow to the Boron Waste Storage Tank.

B. CS-LV-112A, Letdown Divert to BWST is FULL OPEN diverting flow to the Boron Waste Storage Tank.

C. CS-LCV-112A, Letdown Divert to PDT is THROTTLED OPEN diverting flow to the Primary Drain Tank.

D. CS-LCV-112A, Letdown Divert to PDT is FULL OPEN diverting flow to the Primary Drain Tank.

Proposed Answer: C

This question pertains to material associated with multiple common operator misconceptions. There have been operator knowledge issues pertaining to CS-LV-112A and CS-LCV-112A with regard to a) the difference in function between the two valves, b) the human performance error challenge associated with the similarity in valve nomenclature and c)the human performance error challenge associated with the physical system configuration/layout of the components on the control board. These issues contribute to the plausibility of all of the distractors.

C is correct. With all CVCS controls in normal alignment VCT level transmitter LT-185 will throttle open CS-LCV-112A from a range of 75-83% VCT level. The question stem states that VCT level is at 81% so the valve should be in the throttled open position.

A is incorrect but plausible. CS-LV-112A is fed an auto signal from the VCT level control scheme, and would have a throttled open demand, however with the letdown degassifier out of service that flowpath is isolated.

B is incorrect but plausible. CS-LV-112A is fed an auto signal from the VCT level control scheme however the valve would not be full open until level is >83% per LT-185. Additionally, with the letdown degassifier out of service that flowpath is isolated.

D is incorrect but plausible. Letdown flow will be diverting to the PDT via CS-LCV-112A however the valve will be throttled open. With LK-185 setpoint in its normal condition the valve will throttle between 75 and 83% VCT level. CS-LCV-112A would not be full open until level was at 83%.

Technical Reference(s):	1-NHY-503347, CS-TK-1 Level Ctrl Valves Logic Diagram		1-NHY-506275, Chem. Vol and Control Tank TK-1 Control Loop Diagram

Proposed references to be provided to applicants during examination:	None
K/A Topic:	004 Chemical and Volume Control
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.2 to 41.9/ 45.7 to 45.8
Learning Objective:	L8024I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 30	Group #		1		
	K/A #		004 Chemical and Volume Control K4 Knowledge of the CVCS design features and/or interlocks which provide for the following: K4.07 Water Supplies		
	Importance Rating		3.0		

Proposed Question:

Given the following plant conditions:

- A loss of coolant accident occurs.
- RCS pressure is 1780 psig and decreasing.
- All ESF equipment operates as designed.

Based on the given plant conditions when will CS-LCV-112B, 'VCT Outlet Valve' automatically close?

A. ONLY if VCT level decreases below 17% .

B. ONLY if VCT level decreases below 5%.

C. ONLY after CS-LCV-112D, RWST to Charging Pump Suction Valve opens.

D. ONLY after CS-LCV-112E, RWST to Charging Pump Suction Valve opens.

Proposed Answer: C

C is correct. Given the conditions in the stem of the question a Safety Injection signal will have actuated. When the Safety Injection signal actuated CS-LCV-112D and CS-LCV-112E will automatically open to align the charging pumps to the RWST water source. CS-LCV-112B will not close until CS-LCV-112D is open to ensure a suction source to the charging pumps. The 112D/E full open interlock is train specific meaning the CS-LCV-112D interlock is associated with CS-LCV-112B and the CS-LCV-112E interlock is associated with CS-LCV-112C

A is incorrect but plausible. A swapover to the RWST water source does occur based on VCT level however the setpoint is 5% vice 17%.

B is incorrect but plausible. A swapover to the RWST water source does occur based on VCT level @5% however CS-LCV-112D also must be open. Additionally, given the conditions in the question stem the valve would open based on the SI signal vice VCT low level.

D is incorrect but plausible. Given the conditions in the stem of the question a Safety Injection signal will have actuated. When the Safety Injection signal actuated CS-LCV-112D and CS-LCV-112E will automatically open to align the charging pumps to the RWST water source. CS-LCV-112B will not close until CS-LCV-112D is open vice CS-LCV-112E.			
Technical Reference(s):	1-NHY-503341, CS-TK-1 Outlet Isol Valves Logic Diagram		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	004 Chemical and Volume Control		
Question Source:	Bank. Seabrook 2003 Company Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7		
Learning Objective:	L8024I04		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 31	Group #		1		
	K/A #		005 Residual Heat Removal A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: A1.02 RHR flow rate		
	Importance Rating		3.3		

Proposed Question:

Given the following plant conditions:

- A plant shutdown for refueling is in progress.
- Train 'A' Residual Heat Removal is in service providing shutdown cooling.
- RCS pressure is stable at 340 psig.
- RH-FK-618, RHR Train A Flow Control is in automatic maintaining system flow at 3500 gallons per minute.
- RH-HCV-606 is Throttled Open 25% maintaining reactor coolant temperature at 250°F.
- Subsequently RH-FT-618, RHR Loop A Flow fails low.

What change in system configuration will occur?

A. RHR total system flow will increase. Reactor coolant temperature will increase.

B. RHR total system flow will be maintained at 3500 gpm. Reactor coolant temperature will decrease.

C. RHR total system flow will increase. Reactor coolant temperature will decrease.

D. RHR total system flow will be maintained at 3500 gpm. Reactor coolant temperature will increase.

Proposed Answer: A

A is correct. RH-FCV-618, RHR Train A Flow Control Valve is configured in the system such that it bypasses around the heat exchanger/temperature control loop. If RH-FT-618 failed in the low direction then RH-FK-618 would have an input signal which is less than the 3500 gpm auto setpoint. This would cause RH-FCV-618 to fail to the full open position. Total system flow would increase with more flow bypassing the heat exchanger/temperature control loop. This reconfiguration would cause reactor coolant temperature to increase.

B is incorrect but plausible. If RH-HCV-606 were operating in automatic it is plausible that the valve would reconfigure to compensate for the RH-FCV-618 transient. In this case reactor coolant temperature would actually increase. Additionally, RH-HCV-606 is in a modulate mode and its positioning signal is generated by operator adjustment of a dial on the control board.

C is incorrect but plausible. It is true that total system flow would increase as described for the answer A explaination. It is plausible that RCS temperature could decrease if the candidate confused increase in total flow to be through the RHR heat exchanger.

D is incorrect but plausible. If RH-HCV-606 were operating in automatic it is plausible that the valve would reconfigure to compensate for the RH-FCV-618 transient. In this case reactor coolant temperature would actually increase however the positioning signal for RH-HCV-606 signal is generated by operator adjustment of a potentiometer on the control board.

Technical Reference(s):	OS1013.03, Residual Heat Removal Train A Startup and Operation, Section 4.2, RHR Train A Startup From Standby for RCS Cooldown.		1-NHY-503767, RH-Heat Ex 9A&B Outlet Valves Logic Diagram
	1-NHY-503764, RH Pumps Low Flow Recirc Valves Logic Diagram		1-NHY-506650, RH-Pump 8A Control Loop Diagram
	1-NHY-506651, RH-Heat Ex 9A Control Loop Diagram		1-NHY-506652, RH-Heat Ex 9A Bypass Valve Control Loop Diagram

Proposed references to be provided to applicants during examination:	None

K/A Topic:	005 Residual Heat Removal
Question Source:	Modified from bank. Byron 2006 NRC Exam.
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/45.5
Learning Objective:	L8033I07

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 32	Group #		1		
	K/A #		005 Residual Heat Removal 2.2.22 Knowledge of limiting conditions for operation and safety limits.		
	Importance Rating		4.0		

Proposed Question:

Given the following plant conditions:

- The reactor vessel head and upper internals are removed.
- Residual Heat Removal Train 'B' is in operation.
- Residual Heat Removal Train 'A' is in standby.
- Reactor core offload has NOT begun.
- The reactor cavity level is at 23.75 feet above the reactor vessel flange.
- Electrical bus E5 is de-energized due to a bus fault and will be unavailable for several days.

What Technical Specification ACTION, if any, is required?

A. Only one train of Residual Heat Removal is required to be OPERABLE. No Tech. Spec. ACTION is applicable.

B. Immediately initiate corrective action to return the required RHR loops to OPERABLE status as soon as possible.

C. Suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

D. Suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the RCS and immediately initiate corrective action to return the required RHR loops to OPERABLE status.

Proposed Answer: A

A is correct. Tech. Spec. 3.9.8.1 Refueling Operations-Residual Heat Removal and Coolant Recirculation-High Water Level is applicable in Mode 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet. The specification requires that at least one RHR loop shall be OPERABLE and in operation. This specification is applicable for the conditions listed in the stem of the question. The Train 'B' RHR loop is operable and in service so no action is required.

B is incorrect but plausible. This would be the required action if Tech. Spec. 3.9.8.2 Refueling Operations-Residual Heat Removal and Coolant Recirculation-Low Water Level applied. The stem of the question states that the water level is >23 ft.

C is incorrect but plausible. Per Tech. Spech. 3.9.10 this action would be the required if reactor cavity level were less than 23 ft. above the reactor vessel flange. The stem of the question states that the water level is >23 ft.

D is incorrect but plausible. Tech. Spec. 3.9.8.1 Refueling Operations-Residual Heat Removal and Coolant Recirculation-High Water Level is applicable in Mode 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet. The specification requires that at least one RHR loop shall be OPERABLE and in operation. This specification is applicable for the conditions listed in the stem of the question however the action listed in the answers applies if no RHR loop is operable.

Technical Reference(s):	Tech. Spec. 3.9.8.1 Refueling Operations-Residual Heat Removal and Coolant Recirculation-High Water Level		Tech. Spec. 3.9.8.2 Refueling Operations-Residual Heat Removal and Coolant Recirculation-Low Water Level
	Tech. Spec. 3.9.10, Water Level-Reactor Vessel.		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	005 Residual Heat Removal		
Question Source:	Bank. Teb 18709		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.5/43.2/45.2		
Learning Objective:	L8033I14		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 33	Group #		1		
	K/A #		006 Emergency Core Cooling A4 Ability to manually operate and/or monitor in the control room: A4.07 ECCS pumps and valves.		
	Importance Rating		4.4		

Proposed Question:

Given the following plant conditions:
- A Large Break LOCA has occurred.
- All safeguards equipment functioned as designed.
- NO safeguards actuation signals have been RESET.
- The RWST LO-LO level alarm has actuated.

How is ECCS swapover to Cold Leg Recirculation accomplished?

A. CBS-V-8 and CBS-V-14, Containment Recirculation Sump Suction Valves will automatically open. CBS-V-2 and CBS-V-5, Refueling Water Storage Tank Suction Valves will automatically close when the Containment Recirculation Sump Suction Valves are full open.

B. CBS-V-8 and CBS-V-14, Containment Recirculation Sump Suction Valves will automatically open. CBS-V-2 and CBS-V-5, Refueling Water Storage Tank Suction Valves must be manually closed when the Containment Recirculation Sump Suction Valves are full open.

C. CBS-V-8 and CBS-V-14, Containment Recirculation Sump Suction Valves must be manually opened. CBS-V-2 and CBS-V-5, Refueling Water Storage Tank Suction Valves must be manually closed after the Containment Recirculation Sump Suction Valves are fully open.

D. CBS-V-8 and CBS-V-14, Containment Recirculation Sump Suction Valves must be manually opened. CBS-V-2 and CBS-V-5, Refueling Water Storage Tank Suction Valves will automatically close when the Containment Recirculation Sump Suction Valves are full open.

Proposed Answer: B

B is correct. As long as an S signal is present, the containment valves, CBS-V8 and CBS-V14, will auto open. After they are open, S can be reset, and the RWST suctions, CBS-V2 and CBS-V5, may be manually closed from the control room.

A is incorrect but plausible. CBS-V-2 and 5 must be closed but do not auto close.

C is incorrect but plausible. CBS-V-8 and 14 must be opened but will auto open. D is incorrect but plausible. CBS-V-8 and 14 must be opened but will auto open. CBS-V-2 and 5 must be manually closed.			
Technical Reference(s):	1-NHY-503255, CBS-RWS Tank 8 Discharge Valve Logic Diagram		1-NHY-503252, CBS-Cont Sump Isolation Valves Logic Diagram
Proposed references to be provided to applicants during examination:			None
K/A Topic:	006 Emergency Core Cooling		
Question Source:	Bank. Seabrook 2007 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7/45.5 to 45.8		
Learning Objective:	L8035I13, L1203I08		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 34	Group #		1		
	K/A #		007 Pressurizer Relief/Quench Tank A3 Ability to monitor automatic operation of the PRTS, including: A3.01 Components which discharge to the PRT.		
	Importance Rating		2.7		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- B7454 RCS Identified Leak Rate High is in alarm.
- Pressurizer PORV discharge temperature is 103°F
- Pressurizer Relief Valve 115 discharge temperature is 104°F
- Pressurizer Relief Valve 116 discharge temperature is 106°F
- Pressurizer Relief Valve 117 discharge temperature is 108°F
- Pressurizer Relief Tank level is 73% and increasing.
- Pressurizer Relief Tank temperature is 122°F and slowly decreasing.
- Pressurizer Relief Tank pressure is 10 psig and slowly increasing.
- The Pressurizer Relief Tank pump has started and is recirculating the tank through it's heat exchanger.

What would cause these plant conditions?

A. The reactor vessel inner seal is leaking.

B. A Pressurizer PORV or Relief valve is leaking.

C. CS-V-148, Letdown Line 600 psig Relief Valve is leaking.

D. RH-V-13, Train 'A' RHR Discharge Line 600 psig Relief Valve is leaking.

Proposed Answer: C

C is correct. CS-V-148 discharges to the PRT. PRT level is an input to the RCS Identified Leak Rate monitor. The letdown fluid at the relief valve is at approximately 255°F and 350 psig so it would cause a rise in PRT temperature, level and pressure. PRT temperature is slowly decreasing despite the discharge flow from CS-V-148 as the PRT is recirculating through it's heat exchanger.

A is incorrect but plausible. A leak through the reactor vessel inner seal would cause an increase in the identified leak rate however the inner seal leak detection line is routed to the Reactor Coolant

Drain Tank vice the PRT. B is incorrect but plausible. The PORVs and pressurizer relief valve tailpipes are routed to the PRT and would cause an increase in identified leakage. If any of the relieving valves were leaking then there would be an isenthalpic process between the relieving valve and the PRT. The tailpipe temperatures would correspond to the saturation temperature for the given PRT pressure. With PRT pressure at 10 psig the corresponding tailpipe temperatures would be approximately 239°F. D is incorrect but plausible. If the RHR discharge relief valve were leaking it could be caused by a loss of reactor coolant inventory through leaking RHR injection line check valves. The RHR discharge line relief valves are routed to the Primary Drain Tank via the SI test header. Additionally, this distractor is plausible as it could be confused with the RHR loop suction line relief valves which are routed to the Pressurizer Relief Tank.			
Technical Reference(s):	1-NHY-506269, CS Regen HX-E-2 Ltdn Control Loop Diagram		1-NHY-506641, RC Pressurizer Relief Control Loop Diagram
	1-NHY-506643, RC Pressurizer Relief Tank Control Loop Diagram		1-NHY-506621, RC Reactor Vessel Flange Leakoff Control Loop Diagram
	OX1401.02, Form B, Manual RCS Leak Rate		1-NHY-506906, WLD RC Drain Tank Influent Drains Temperature Control Loop Diagram
	PID-1-RH-20662, Residual Heat Removal System Train A Detail		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	007 Pressurizer Relief/Quench Tank
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.5
Learning Objective:	L8022I01, L8024I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 35	Group #		1		
	K/A #		008 Component Cooling Water K2 Knowledge of the bus power supplies to the following: K2.02. CCW pump, including emergency backup.		
	Importance Rating		3.0		

Proposed Question:

Given the following plant conditions:

- The Reactor Operator is in the process of swapping running Primary Component Cooling Water (PCCW) pumps in the 'A' PCCW Loop.
- Both 'A' & 'C' PCCW pumps are running when a loss of off-site power occurs.
- All systems function as designed.

What will be the status of Train "A" PCCW pumps upon completion of emergency power sequencing?

A. Both 'A' and 'C' PCCW pumps tripped.

B. Both 'A' and 'C' PCCW pumps running.

C. Only the 'A' PCCW pump is running and the 'C' PCCW pump is tripped.

D. Only the 'C' PCCW pump is running and the 'A' PCCW pump is tripped.

Proposed Answer: D

D is correct. The "C" PCCW pump is electrically the preferred pump. In the event of a loss of off-site power and subsequent re-energization of Bus 5 from the Train A Emergency Diesel Generator, the 'C' pump will start. The purpose of this configuration is to prevent the 'A' EDG from overloading during initial starting and sequencing. The 'A' pump power source is locked out until RMO is reset by the operator.

A is incorrect but plausible. It is plausible that both PCCW pumps could be locked out by RMO to prevent overloading the emergency diesel, however PCCW is a sequenced load with the 'C' pump being the preferred pump.

B is incorrect but plausible. It is plausible that both pumps would restart as their control switches

would be in the Normal After Start position, however the power source for the pumps is designed such that the 'C' pump is preferred. C is incorrect but plausible. It is correct that one pump is preferred and one is locked out by RMO however the preferred pump is 'C' vice 'A'.			
Technical Reference(s):	1-NHY-310895 Sh-A58b, PCCW Loop A Pump 1-P-11A Close Schematic		1-NHY-310895 Sh-A58c, PCCW Loop A Pump 1-P-11A Trip Schematic
	1-NHY-310895 Sh-A59b, PCCW Loop A Pump 1-P-11C Close Schematic		1-NHY-310895 Sh-A59c, PCCW Loop A Pump 1-P-11C Close Schematic
Proposed references to be provided to applicants during examination:			None
K/A Topic:	008 Component Cooling Water		
Question Source:	Bank. Seabrook 2005 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7		
Learning Objective:	L8036I06		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 36	Group #		1		
	K/A #		010 Pressurizer Pressure Control K3 Knowledge of the effect that a loss or malfunction of the PZR PCS will have on the following: K3.01 RCS		
	Importance Rating		3.8		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- All systems are in normal alignment.
- Tavg is on program and stable.
- Pressurizer level is on program and stable.
- Pressurizer pressure is 2228 psig and slowly decreasing.

What event has occurred?

A. Pressurizer pressure instrument PT-458 has failed high.
B. RC-PCV-455B, Pressurizer Spray Valve has failed to 10% open.
C. RC-PCV-456A, Pressurizer PORV ‘A’ has failed open.
D. RC-PK-455A, Pressurizer Master Pressure Controller output signal has failed low.

Proposed Answer: B

B is correct. A partially open spray valve will cause a condensing action of the pressurizer steam bubble and initiate a slow pressurizer pressure decrease transient.

A is incorrect but plausible. It is plausible that a high failure of PT-458 would cause a PORV to open however a single pressure instrument failure will not open a PORV.

C is incorrect but plausible. A pressurizer PORV failing open would cause pressurizer pressure to decrease however the rate of depressurization would be much greater than the condition listed in the question stem. Additionally, an open PORV represents a steam space LOCA which would be accompanied by an increase in pressurizer level.

D is incorrect but plausible. A low failure of the master pressure controller would cause an increase

in pressurizer pressure as this failure would cause the control and backup heaters to energize.			
Technical Reference(s):	1-NHY-509026, Pressurizer Pressure Control Block Diagram		1-NHY-509027, Pressurizer Level Control Process Control Block Diagram

Proposed references to be provided to applicants during examination:	None
K/A Topic:	010 Pressurizer Pressure Control
Question Source:	Bank. Edited from 2006 Byron NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.6
Learning Objective:	L8027I06, L8027I10, L1406I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 37	Group #		1		
	K/A #		010 Pressurizer Pressure Control A4 Ability to manually operate and/or monitor in the control room: A4.02 PZR heaters.		
	Importance Rating		3.6		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- The Pressurizer Master Pressure Controller is in automatic.
- All Pressurizer Heaters are in automatic.
- The Control Group Heaters are NOT ENERGIZED.
- All backup heaters are ENERGIZED.

What conditions in the pressurizer support the plant conditions listed above?

A. Pressurizer pressure: 2150 psig Pressurizer level:10%.
B. Pressurizer pressure: 2150 psig Pressurizer level:61%.
C. Pressurizer pressure: 2260 psig Pressurizer level:10%.
D. Pressurizer pressure: 2260 psig Pressurizer level:67%.

Proposed Answer: D

D is correct. With pressurizer pressure at 2260 psig all of the heaters (control and backup) would normally be de-energized as this is above the controller output signal range calling for pressurizer heaters. Pressurizer level is 67% which is 7% higher than the 100% power setpoint of 60%. The pressurizer backup heaters automatically energize when pressurizer level is >5% above setpoint.

A is incorrect but plausible. At 2150 psig the output of the master pressure controller would call for energization of all heaters. Pressurizer level is at 10% which is below the heater trip setpoint of 17%. It is plausible that the Control Group heaters would be de-energized due to the stated pressurizer level however all of the heaters should be de-energized.

B is incorrect but plausible. At 2150 psig the output of the master pressure controller would call for energization of all heaters. It is true that all backup heaters would be energized however the control group heaters would be energized as well.

C is incorrect but plausible. It is plausible for backup heaters to be energized at a pressure of 2260 psig as would be the case during a scenario when the operators force pressurizer sprays however this is done by manually energizing sets of backup heaters. Additionally, pressurizer level is at 10% which is below the heater trip setpoint of 17%. It is plausible that the Control Group heaters would be de-energized due to the stated pressurizer level however all of the heaters should be de-energized.

Technical Reference(s):	1-NHY-310822		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	010 Pressurizer Pressure Control
Question Source:	Bank. TEB 16294
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.5 to 45.8
Learning Objective:	L8027I05, L8027I06, L8027I08,

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 38	Group #		1		
	K/A #		012 Reactor Protection K4 Knowledge of the RPS design feature(s) and/or interlocks(S) which provide for the following: K4.06 Automatic or manual enable/disable of reactor trips.		
	Importance Rating		3.2		

Proposed Question:

Given the following plant conditions:

- A plant startup is in progress.
- Five minutes ago the crew noted the following conditions:
  - Reactor power indicated 20% and increasing on all four Power Range Nuclear Instrumentation channels, and
  - Turbine impulse pressure indicated 128 psig and increasing.

What is the status of C-20, ATWS Mitigation permissive and why?

A. ATWS Mitigation is armed. It was armed when turbine impulse pressure exceeded 125 psig.

B. ATWS Mitigation is not armed. It will arm when turbine impulse pressure exceeds 125 psig for greater than 6 minutes.

C. ATWS Mitigation is armed. It was armed by the Power Range Nuclear Instrumentation channels when 2 of 4 channels exceeded 20% power.

D. ATWS Mitigation is not armed. It will arm when 2 of 4 Power Range Nuclear Instrumentation channels exceed 20% power for greater than 6 minutes.

Proposed Answer: A

A is correct. The Reactor Protection System C-20, AMSAC Control Permissive will arm when either turbine first stage pressure instrument (PT-505 or 506) exceeds 125 psig (20% equivalent power).

B is incorrect but plausible. There is a 6 minute time delay associated with the C-20 arming circuit however it is a 6 minute dropout where the C-20 signal will disable if both PT-505 and 506 drop below 125 psig for greater then 6 minutes.

C is incorrect but plausible. It is true that C-20 arms when power exceeds 20% however this is a derived turbine power level from PT-505 and PT0506 vice a power level signal from Nuclear Instrumentation.

D is incorrect but plausible. It is true that C-20 arms when power exceeds 20% however this is a derived turbine power level from PT-505 and PT0506 vice a power level signal from Nuclear Instrumentation. Additionally there is a 6 minute time delay associated with the C-20 arming circuit however it is a 6 minute dropout where the C-20 signal will disable if both PT-505 and 506 drop below 125 psig for greater then 6 minutes.

Technical Reference(s):	OS1235.05, Turbine Impulse Pressure PT 505 or PT 506 Instrument Failure, Step 5.		1-NHY-593004

Proposed references to be provided to applicants during examination:	None
K/A Topic:	012 Reactor Protection
Question Source:	Bank
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.7
Learning Objective:	L8056I23

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 39	Group #		1		
	K/A #		012 Reactor Protection K6 Knowledge of the effect that a loss or malfunction of the following will have on the RPS: K6.10 Permissive Circuits		
	Importance Rating		3.3		

Proposed Question:

Given the following plant conditions:

- The plant is initially at 100% power.
- An inadvertent Safety Injection occurs.
- The Train 'B' Reactor Trip Breaker will not open.
- All other systems and equipment respond as expected.
- Per E-0, "Reactor Trip or Safety Injection" the crew has reset the Safety Injection signal on both trains.

What is the status of the Safety Injection circuitry?

A. Both trains of SI are reset. SI automatic actuation is blocked on both trains.
B. Only Train 'A' SI is reset. Only Train 'A' SI automatic actuation is blocked.
C. Both trains of SI are reset. Only Train 'A' SI automatic actuation is blocked.
D. Neither trains SI signal is reset. SI automatic actuation is not blocked on either train.

Proposed Answer: C

C is correct. A train specific P-4 signal allows for blocking of further automatic SI actuations once that train's SI signal is reset. Failure of Reactor Trip Breaker 'B' to open will inhibit the ability to block a further automatic SI on Train 'B' once SI is reset. The SI reset is still functional since the reset signal comes from dual train switches that do not have a P-4 interface.

A is incorrect but plausible. Both trains of SI will reset as the SI switches are dual train and have no P-4 interface. It is plausible that the auto SI signals would be blocked on both trains if the blocking signal were dual train however that signal is train specific with a P-4 signal being generated from the specific train related reactor trip breaker actuation.

B is incorrect but plausible. It is plausible that both the SI reset and auto SI block would both require a P-4 signal input however both trains of SI will reset as the SI switches are dual train and have no P-4 interface.

D is incorrect but plausible. It is plausible that the P-4 signal would be dual train with both train related P-4 signals required to be reset.			
Technical Reference(s):	1-NHY-509042, Reactor Trip Signals w/ Functional Diagrams		1-NHY-509048, Safeguard Actuation Signals w/ Functional Diagrams
Proposed references to be provided to applicants during examination:	None		
K/A Topic:	012 Reactor Protection		
Question Source:	Bank. Seabrook 2003 Company Exam		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.7/45.7		
Learning Objective:	L8056I29		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 40	Group #		1		
	K/A #		013 Engineered Safety Features Actuation K5 Knowledge of the operational implications of the following concepts as they apply to the ESFAS: K5.02 Safety system logic and reliability.		
	Importance Rating		2.9		

Proposed Question:

Given the following plant conditions:

- The plant has sustained a Steam Line Break.
- The reactor has tripped.
- The following indications are noted:
  - SG A pressure - 700 psig and slowly DECREASING
  - SG B pressure - 600 psig and steadily DECREASING
  - SG C pressure - 850 psig and STABLE
  - SG D pressure - 850 psig and STABLE
  - RCS pressure - 1880 psig and DECREASING
  - Containment pressure - 8 psig and INCREASING
- All safeguards systems have functioned as designed.
- NO additional actions have been taken.

What ESF Actuations have occurred?

A. Safety Injection, Containment Isolation Phase A ONLY.

B. Safety Injection, Containment Isolation Phase A, Containment Phase B, and Main Steamline Isolation ONLY.

C. Safety Injection, Containment Isolation Phase A, Main Steamline Isolation, and EFW Actuation ONLY.

D. Safety Injection, Containment Isolation phase A, Main Steamline Isolation, EFW Actuation, and Containment Spray Actuation/ Phase B.

Proposed Answer:	C

C is correct. Plant conditions require actuation of containment HI-1 (SI, CIS-A) and HI-2 (MSLIS) (4.3 psig). Steam pressure has not dropped to the Steam Line pressure MSLIS setpoint, but containment pressure has increased above HI-2. An EFW actuation occurs on an SI signal.

A is incorrect but plausible. A Main Steam Isolation and EFW Actuation also occurs.

B is incorrect but plausible. An EFW actuation also occurs. A Phase B actuation does not occur because the setpoint is 18 psig containment pressure.

D is incorrect but plausible. A CBS actuation and Phase B actuation do not occur because the setpoint is 18 psig containment pressure.

Technical Reference(s):	Westinghouse PLS, Seabrook Station, pgs 10-11.		Westinghouse Functional diagram 509048.

Proposed references to be provided to applicants during examination:	None
K/A Topic:	013 Engineered Safety Features Actuation
Question Source:	Bank. Seabrook 2007 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/45.7
Learning Objective:	L8057I08

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 41	Group #		1		
	K/A #		013 Engineered Safety Features Actuation K6 Knowledge of the effect that a loss or malfunction of the following will have on the ESFAS: K6.01 Sensors and detectors.		
	Importance Rating		2.7		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- All plant systems are in normal alignment.
- Containment pressure channel PT-934 is inoperable.
- In accordance with Technical Specifications the following actions have been taken:
  - ➢ The associated Containment Hi-1 bistable has been placed in the TRIPPED condition.
  - ➢ The associated Containment Hi-2 bistable has been placed in the TRIPPED condition.
  - ➢ The associated Containment Hi-3 bistable has been placed in BYPASS.

What are the remaining coincidences required for automatic ESF actuations?

A. Safety Injection Actuation: 2 of 3 channels
   Main Steam Isolation: 2 of 3 channels
   Containment Spray Actuation: 2 of 3 channels

B. Safety Injection Actuation: 1 of 3 channels
   Main Steam Isolation: 1 of 3 channels
   Containment Spray Actuation: 1 of 3 channels

C. Safety Injection Actuation: 1 of 2 channels
   Main Steam Isolation: 1 of 2 channels
   Containment Spray Actuation: 1 of 2 channels

D. Safety Injection Actuation: 1 of 2 channels
   Main Steam Isolation: 1 of 2 channels
   Containment Spray Actuation: 2 of 3 channels

Proposed Answer:	D

D is correct. The Hi-1 Containment Pressure signal utilizes 3 channels. A 2/3 channel coincidence will initiate an automatic Safety Injection. If 1 of the 3 channels has been placed in the trip condition then the remaining logic is 1 of 2.

The Containment Hi-2 signal utilizes 3 channels. A 2/3 channel coincidence will initiate an automatic Main Steam Isolation. If 1 of the 3 channels has been placed in the trip condition then the remaining logic is 1 of 2.

The Containment Hi-3 signal utilizes 4 channels. A 2/4 channel coincidence will initiate an automatic Containment Spray Actuation. In this case the inoperable channel was placed in bypass vice tripped so 2 channels are still required to initiate an automatic signal. The remaining coincidence is 2/3 channels.

A is incorrect but plausible. A 2/3 coincidence would be required if the inoperable Hi-1 channel were placed in bypass vice a tripped condition. A 2/3 coincidence would be required if the inoperable Hi-2 channel were placed in bypass vice a tripped condition.

The Containment Hi-3 signal utilizes 4 channels. A 2/4 channel coincidence will initiate an automatic Containment Spray Actuation. In this case the inoperable channel was placed in bypass vice tripped so it is correct that 2 channels are still required to initiate an automatic signal. The remaining coincidence is 2/3 channels.

B is incorrect but plausible. If the Hi-1 signal were generated from a 2/4 logic then it would be correct that a 2/3 remaining logic would be required. This is plausible as a multitude actuation signals and reactor trip signals utilize a 2/4 logic (Pressurizer pressure, pressurizer level, etc.). The same would be true for a Main Steam Isolation if the Hi-2 signal utilized a 2/4 logic. Additionally, a Hi-3 1/3 logic would be required if the Hi-3 instrument were placed in trip vice bypass.

C is incorrect but plausible. It is true that if 1 of the 3 Hi-1 channels has been placed in the trip condition then the remaining Safety Injection logic would be 1 of 2. It is true that if 1 of the 3 Hi-2 channels has been placed in the trip condition then the remaining Main Steam Isolation logic would be 1 of 2. If the Hi-3 channel utilized a 2/3 logic and the inoperable channel had been placed in trip vice bypass then the remaining Hi-3 CBS actuation logic would be 1 of 2.

Technical Reference(s):	1-NHY-509048, Safeguard Actuation Signals w/Functional Diagrams		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	013 Engineered Safety Features Actuation

Question Source:	Modified from bank. Braidwood 2007 NRC Exam			

Question Cognitive Level:	Higher: Comprehension/Analysis		

10 CFR Part 55 Content:	41.7/45.5 to 45.8	

Learning Objective:	L8057I10, L8057I08

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 42	Group #		1		
	K/A #		022 Containment Cooling K4 Knowledge of the CCS design features(s) and/or interlock(s) which provide for the following: K4.04 Cooling of control rod drive motors.		
	Importance Rating		2.8		

Proposed Question:

Given the following plant conditions:

- The plant was initially at 100% power.
- A LOCA is in progress.
- Reactor Trip and Safety Injection have occurred.
- All safeguards systems are functioning as designed.
- Containment Pressure is 24 psig and trending down slowly.

Which of the following describes the status of Containment Cooling Systems?

A. Control Rod Drive Mechanism Cooling Fans are RUNNING, Containment Structure Cooling Fans are RUNNING, Containment Recirculation Fans are operating in the FILTER MODE.

B. Control Rod Drive Mechanism Cooling Fans are TRIPPED, Containment Structure Cooling Fans are TRIPPED, Containment Recirculation Fans are operating in the FILTER MODE.

C. Control Rod Drive Mechanism Cooling Fans are RUNNING, Containment Structure Cooling Fans are RUNNING, Containment Recirculation Fans are operating in the RECIRC MODE.

D. Control Rod Drive Mechanism Cooling Fans are TRIPPED, Containment Structure Cooling Fans are TRIPPED, Containment Recirculation Fans are operating in the RECIRC MODE.

Proposed Answer: D

D is correct. Given the conditions listed a ‘P’ signal (Hi-3, 18 psig containment pressure) will have occurred. A ‘P’ signal will trip the Control Rod Drive Mechanism Fans, trip the Containment Structure Cooling Fans (indirectly due to a loss of PCCW flow) and will cause the Containment Recirculation Fan subsystem to run in the recirculation mode.

A is incorrect but plausible. It is plausible that all containment subsystem cooling would remain in service during an accident condition to aid in maintaining containment integrity however this is not part of the subsystem design. Additionally, it is plausible that the containment structure recirculation fans would run in the filter mode on a 'P' signal so as to filter post accident products, however the subsystem actually runs in the recirc mode in order to recirculate the containment atmosphere to control hydrogen concentration.

B is incorrect but plausible. It is plausible that the containment structure recirculation fans would run in the filter mode on a 'P' signal so as to filter post accident products, however the subsystem actually runs in the recirc mode in order to recirculate the containment atmosphere to control hydrogen concentration.

C is incorrect but plausible. A 'P' signal will cause the Containment Recirculation Fan subsystem to run in the recirculation mode. Additionally, it is plausible that all containment subsystem cooling would remain in service during an accident condition to aid in maintaining containment integrity however this is not part of the subsystem design.

Technical Reference(s):	1-NHY-503201, CAH-Containment Structure Cooling Unit Fans Logic Diagram		1-NHY-503202, CAH-CRDM Cooling Unit Fans Logic Diagram
	1-NHY-503204, CAH-Containment Structure Recirc Filter Fan Logic Diagram		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	022 Containment Cooling
Question Source:	Bank
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7
Learning Objective:	L8038I04

Seabrook Station 2010 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 43	Group #		1		
	K/A #		026 Containment Spray A3 Ability to monitor automatic operation of the CCS, including: A3.01 Pump starts and correct MOV positioning.		
	Importance Rating		4.3		

Proposed Question:

Given the following plant conditions:

- A LOCA has occurred.
- Safety Injection is actuated.
- Containment pressure is 19 psig, slowly lowering.
- The PSO notes the white light associated with the automatic operation of Containment Sump Isolation valve CBS-V8 is lit.

What is the significance of the white light being lit?

A. A 'P' signal is present enabling CBS-V-8 to automatically open when the RWST EMPTY alarm actuates.

B. An 'SI' signal is present enabling CBS-V-8 to be manually opened when the RWST EMPTY alarm actuates.

C. An 'SI' signal is present enabling CBS-V8 to automatically open when the RWST LEVEL LO-LO alarm actuates.

D. A 'P' signal is present enabling CBS-V-8 to automatically open when the RWST LEVEL LO-LO alarm actuates.

Proposed Answer: C

C is correct. An 'SI' signal combined with 2/4 RWST level channels reaching the LO LO level setpoint (120,478 gallons) will initiate an ECCS/CBS recirc signal. The ECCS/CBS recirc signal will send an automatic opening signal to CBS-V-8. As soon as the 'SI' signal is present the white light associated with CBS-V-8 will illuminate.

A is incorrect but plausible. The light does indicate that CBS-V-8 is armed for automatic opening, however the arming comes from an "SI' signal. Additionally, the valve will automatically open however this occurs at the LO LO level of 120,478 gallons vice when the RWST EMPTY alarm actuates.

B is incorrect but plausible. It is correct that the light is illuminated when an 'SI' signal is present. The light signifies that CBS-V-8 will automatically open vice be manually open by the operator. The semi-automatic ECCS swapover to recirculation presents a series of operator misconceptions as to which valves automatically align and which ones must be manipulated manually. It is conceivable that there would be a control scheme allowing for manual operation of CBS-V-8 in the event that the RWST EMPTY alarm acutuate for the purpose of protecting running ECCS pumps from loss of suction.

D is incorrect but plausible. The light does indicate that CBS-V-8 is armed for automatic opening, however the arming comes from an "SI' signal. It is true that the valve will automatically open when RWST level decreases to 120,478 gallons.

Technical Reference(s):	1-NHY-503252, CBS-Containment Sump Isolation Valves Logic Diagram		1-NHY-503258, CBS-ECCS/Spray Recirc Signal Generation Logic Diagram

Proposed references to be provided to applicants during examination:	None

K/A Topic:	026 Containment Spray
Question Source:	Bank. Teb 23099
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.7/45.5
Learning Objective:	L8035I13, L8035I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 44	Group #		1		
	K/A #		039 Main and Reheat Steam K4 Knowledge of the MRSS design features and/or interlocks which provide for the following: K4.05 Automatic isolation of steam line.		
	Importance Rating		3.7		

Proposed Question:

Given the following plant conditions:

- The plant is initially at 100% power.
- A steam line break occurs inside containment from the 'B' Steam Generator.
- The 'B' Steam Generator completely depressurizes.
- Containment pressure peaks at 16 psig.
- All Main Steam Isolation Valves have closed.

Which of the following directly caused the Main Steam Isolation Valves to close?

A. Phase B (P) Signal.

B. Safety Injection signal.

C. Low Steam Line Pressure.

D. High Steam Pressure Rate.

Proposed Answer: C

C is correct. Given the conditions in the stem of the question an automatic Main Steam Isolation signal could have occurred from either a) Containment Hi-2 (4.3 psig) or b) Low Steam Line Pressure (585 psig, rate compensated, 2/3 transmitters on any Steam Generator). A Containment Hi-2 signal is not listed as one of the answers so the correct answer is Low Steam Line Pressure.

A is incorrect but plausible. A containment Phase B signal does cause automatic closure of various containment penetrations however it does not close the MSIV's. Additionally, the question stem states that containment pressure peaked at 16 psig which is less than the Phase B setpoint of 18 psig.

B is incorrect but plausible. It is a common operator misconception that a Safety Injection signal generated a Main Steam Isolation signal. Given the conditions in the stem the SI signal and the Main Steam Isolation signal could have both occurred when containment pressure reached 4.3 psig

however the signal to isolate the MSIV's is separate from the SI signal. D is incorrect but plausible. It is true that there is a High Steam Pressure Rate isolation signal however that signal is only active when the plant is below the P-11 (1950 psig) setpoint and the SI signal has been manually blocked.			
Technical Reference(s):	1-NHY-509047, sheet 2, FW/STM Gen Trip Signals w/ Functional Diagram		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	039 Main and Reheat Steam		
Question Source:	Modified from bank. Diablo Canyon 2009 NRC Exam		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.7		
Learning Objective:	L8041I09		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 45	Group #		1		
	K/A #		059 Main Feedwater K1 Knowledge of the physical connections and/or cause effect relationships between the MFW and the following systems: K1.04 S/Gs water level control system.		
	Importance Rating		3.4		

Proposed Question:

What is the functional relationship between the Main Feedwater Pumps and the Main Feedwater Regulating Valves when plant power is increasing from 50% to 100%?

A. The Main Feedwater Pumps maintain a constant differential pressure across the Main Feedwater Regulating Valves. The Main Feedwater Regulating Valves throttle to maintain a variable water level in the Steam Generators.

B. The Main Feedwater Pumps maintain a power dependant variable differential pressure across the Main Feedwater Regulating Valves. The Main Feedwater Regulating Valves throttle to maintain a constant water level in the Steam Generators.

C. The Main Feedwater Pumps maintain a constant differential pressure across the Main Feedwater Regulating Valves. The Main Feedwater Regulating Valves throttle to maintain a constant water level in the Steam Generators.

D. The Main Feedwater Pumps maintain a power dependant variable differential pressure across the Main Feedwater Regulating Valves. The Main Feedwater Regulating Valves throttle to maintain a variable water level in the Steam Generators.

Proposed Answer: B

B is correct. The main feedwater pump master controller is programmed to maintain a power dependant variable differential pressure across the main feedwater regulating valves. The setpoint ramps from 80 to 135 psid over a range of 20 to 100% power. The ramped differential pressure program is designed to allow the feedwater regulating valves to throttle near a 50% open position. The main feed regulating valve controllers are programmed to maintain a constant 50% level in the steam generators.

A is incorrect but plausible. It is plausible that the main feedwater pumps could be programmed to maintain a constant differential pressure across the feedwater regulating valves such that the regulating valves throttle accordingly to maintain constant generator level. The feedwater pumps are programmed to maintain a ramped differential pressure across the feedwater regulating valves.

Additionally, the controller maintains a constant generator level.

C is incorrect but plausible. It is plausible that the main feedwater pumps could be programmed to maintain a constant differential pressure across the feedwater regulating valves such that the regulating valves throttle accordingly to maintain constant generator level. The feedwater pumps are programmed to maintain a ramped differential pressure across the feedwater regulating valves. It is correct that the steam generator level setpoint is constant at 50%.

D is incorrect but plausible. It is true that the main feedwater pump master controller is programmed to maintain a power dependant variable differential pressure across the main feedwater regulating valves. Additionally, it is plausible that the steam generator level setpoint could vary to maintain a constant mass of secondary coolant in the steam generators however the level setpoint is constant at 50%.

Technical Reference(s):	Secondary Technical Data Book, MISC-10, Main Feedwater Pump DP		1-NHY-509025, Steam Dump and Feed Pump Speed Control Process Control Block Diagram
	1-NHY-509033, Steam Generator Level Control Process Control Block Diagram		Westinghouse PLS, page 24, item 6, Feedwater Control
Proposed references to be provided to applicants during examination:	None		
K/A Topic:	059 Main Feedwater		
Question Source:	Bank. Seabrook 2005 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.2 to 41.9 /45.7 to 45.8		
Learning Objective:	L8046I02, L8046I03, L8046I04, L8046I05		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 46	Group #		1		
	K/A #		061 Auxiliary/Emergency Feedwater K2 Knowledge of the bus power supplies to the following: K2.01 AFW system MOV's.		
	Importance Rating		3.2		

Proposed Question:

Given the following plant conditions:

- A steam break in the 'C' steam generator resulted in a reactor trip and Safety Injection.
- All plant systems functioned as designed.
- The control room crew restored an adequate heat sink with the following control switch (CS) alignment for the EFW flow control valves:

Steam Generator A
CS-4214-A1 Throttled Full Closed
CS-4214-B1 Auto-Full Open
Steam Generator B
CS-4224-A1 Auto-Full Open
CS-4224-B1 Throttled Full Closed
Steam Generator C
CS-4234-A1 Auto-Full Closed
CS-4234-B1 Auto-Full Closed
Steam Generator D
CS-4244-A1 Auto-Full Open
CS-4244-B1 Throttled Full Closed

- Subsequently power to MCC-515 is lost

What is the effect on the control room operator's ability to control steam generator level, if any?

A. No effect since the loss of MCC affects only one of the two flow control valves to each steam generator.
B. The operator will be unable to initiate flow to the 'A' and 'D' steam generators. Local valve operation would need to be coordinated with an NSO.
C. The operator will be unable to initiate flow to the 'B' and 'D' steam generators. Local valve operation would need to be coordinated with an NSO.

D. The operator will be unable to initiate flow to the 'A' steam generator. Local valve operation would need to be coordinated with an NSO.		
Proposed Answer:	D	

D is correct. 'A' Steam Generator throttle valve FW-FCV-4214A, which is controlled by control switch CS-4214-A1, is powered from electrical train A via MCC-515. The valve is throttled full closed. The control board operator would not be able to reopen the valve with the control switch. An NSO would have to manually operate the valve in order to reestablish flow.

A is incorrect but plausible. It is true that each EFW line has a Train A and Train B electrically supplied valve and that one valve is affected in each line however any SG that has a Train 'A' valve throttled full closed would be affected.

B is incorrect but plausible. It is true that the control board operator would not be able to reopen the throttled closed valve on Steam Generator 'A' with the control switch however the throttled closed valve for Steam Generator 'D' is powered from Train B (MCC-615) so the valve could be reopened with the control switch.

C is incorrect but plausible. It is true that each EFW line has a Train A and Train B electrically supplied valve and that one valve is affected in each line however the closed valves on the 'B' and 'D' steam generators are powered from Train B and could be opened.

Technical Reference(s):	OS1036.01, Aligning The Emergency Feedwater System For Automatic Initiation, Attachment A, Emergency Feedwater System Lineup, pages 24-26, Electrical Lineup.		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	061 Auxiliary/Emergency Feedwater
Question Source:	Modified. Modified from Seabrook 2007 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7
Learning Objective:	L8045I01, L8045I02

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 47	Group #		1		
	K/A #		061 Auxiliary/Emergency Feedwater A3 Ability to monitor automatic operation of the AFW, including: A3.02 RCS cooldown during AFW operations.		
	Importance Rating		4.0		

Proposed Question:

Given the following plant conditions:

- The plant was initially at 100% power.
- The reactor has tripped due to Main Turbine Trip.
- The crew entered E-0, Reactor Trip or Safety Injection.
- Reactor Coolant temperature is 551°F and slowly decreasing.
- The crew is taking action to throttle Emergency Feedwater flow to stop the RCS cooldown.
- Subsequently the 'B' Main Steam Line breaks inside containment.

How will the Emergency Feedwater system respond?

A. The EFW flow control valves on the 'B' Steam Generator will auto close when flow reaches 525 gpm. EFW flow will be limited to 525 gpm on the intact Steam Generators by a venturi in the EFW line.
B. The EFW flow control valves on the 'B' Steam Generator will auto close when flow reaches 750 gpm. EFW flow will be limited to 750 gpm on the intact Steam Generators by a venturi in the EFW line.
C. The EFW flow control valves on the 'B' Steam Generator will auto close when flow reaches 525 gpm. EFW flow will be limited to 750 gpm on any Steam Generator by a venturi in the EFW line.
D. The EFW flow control valves on the 'B' Steam Generator will auto close when flow reaches 525 gpm. The EFW control valves for the intact Steam Generators will auto close if flow reaches 750 gpm.

Proposed Answer: C

C is correct. The EFW flow control valves are designed to automatically close if flow increases to >525 gpm on that specific EFW line. All of the Steam Generator EFW lines contain an internal venturi that restricts flow through that specific line to 750 gpm.

A is incorrect but plausible. It is true that the EFW valves have an automatic closing setpoint at

525 gpm. It is also true that there is a flow limiting venture on all of the EFW lines however they are designed to limit flow to 750 gpm vice 525 gpm.

B is incorrect but plausible. It is true that there is an automatic setpoint for EFW valve closure but it is 525 gpm not 750 gpm. It is also true that there is a flow limiting venture on all of the EFW lines designed to limit flow to 750 gpm.

D is incorrect but plausible. It is true that the EFW valves on the affected generator will auto close when flow reaches 525 gpm. The flow isolation signal is designed such that it does not cascade and isolate all steam generators when their flow exceeds 525 gpm. The cascading feature of the EFW isolation circuit is a common operator misconception. It is true that other EFW lines would auto isolate, however this would happen if the previously isolated steam generator EFW valves were reopened without breaking the signal seal in feature for the isolation signal to the other EFW isolation valves. This feature does not include any additional or secondary isolation setpoint at 750 gpm.

Technical Reference(s):	1-NHY-504152, FW Emergency Valves Logic Diagram		1-NHY-506497, FW-EFW-P-37A Control Loop Diagram

Proposed references to be provided to applicants during examination:	None

K/A Topic:	061 Auxiliary/Emergency Feedwater
Question Source:	Modified. Seabrook 2005 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.7/45.5
Learning Objective:	L8045I07

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 48	Group #		1		
	K/A #		062 AC Electrical Distribution K1 Knowledge of the physical connections and/or cause-effect relationships between the ac distribution system and the following systems: K1.02 ED/G		
	Importance Rating		4.1		

Proposed Question:

Which of the following describes the operation of the Emergency Bus <u>second</u> <u>level</u> undervoltage protection scheme?

A. There are two normally de-energized undervoltage relays. When Bus voltage drops below 25% of nominal, as sensed by either relay they energize, initiating auto closure of the RAT supply breaker.

B. There are two normally energized undervoltage relays. When either relay senses Bus voltage less than 95% of nominal for 1.2 seconds (RAT available), they initiate a sequence of load stripping and subsequent Bus reenergization by the EDG.

C. There are two normally de-energized undervoltage relays. When both relays sense Bus voltage less than 70% of nominal for 1.2 seconds (RAT available), it initiates a sequence of load stripping and subsequent Bus reenergization by the EDG.

D. There are two normally energized undervoltage relays. When both relays sense Bus voltage less than 95% of nominal coincident with an SI existing for greater than 10 seconds, they initiate a sequence of load stripping and subsequent Bus reenergization by the EDG.

Proposed Answer: D

D is correct. Second level undervoltage protection for emergency busses is actuated when 2/2 UV relays sense <95% nominal bus voltage w/ SI signal present for >10 sec.

A is incorrect but plausible. This is the description of an associated bus relay however it is the relay associated with a bus auto swap to the RAT transformer.

B is incorrect but plausible. This distractor describes the first level undervoltage protection logic with the second level undervoltage setpoint.

C is incorrect but plausible. This distractor describes the first level undervoltage protection.			
Technical Reference(s):	1-NHY-310102, Sheet A53a, 4160 Bus 1-E5 PT's Three Line Diagram		
Proposed references to be provided to applicants during examination:		None	
K/A Topic:	062 AC Electrical Distribution		
Question Source:	Bank. Teb 30082		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.2 to 41.9		
Learning Objective:	L8013I13		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 49	Group #		1		
	K/A #		063 DC Electrical Distribution K4 Knowledge of DC electrical system design feature(s) and/or interlock(s) which provide for the following: K4.02 Breaker interlocks, permissives, bypasses and cross-ties.		
	Importance Rating		2.9		

Proposed Question:

The operating procedure for transferring a Vital 125VDC bus to its Alternate Battery Supply utilizes a Kirk Key interlock device to ensure that the busses Normal Battery Supply Breaker is opened before the Alternate Battery Supply Breaker is closed. What is the reason for this interlock design feature?

A. To prevent battery banks from separate Vital 125VDC electrical trains from operating in parallel.

B. To prevent the battery chargers on the same Vital 125VDC electrical train from operating in parallel.

C. To prevent both battery banks on the same Vital 125VDC electrical train from being placed in parallel.

D. To prevent a battery charger on a Vital 125VDC electrical train from being paralleled to the portable battery charger.

Proposed Answer: C

C is correct. The normal battery supply breaker is opened before the alternate battery supply breaker is closed to prevent excessive current conditions associated with parallel configuration of both battery banks on the same train.

A is incorrect but plausible. It is plausible that a battery bank from a separate vital DC bus could physically be used as an alternate battery supply for the other bus however the Vital 125VDC system is designed with two separate safety related trains that are physically detached from each other.

B is incorrect but plausible. It is conceivable that the Kirk Key interlock arrangement would be in place to protect the battery chargers from parallel operation however this is not the case. In fact the operating procedure for transferring a DC bus to it's alternate battery supply includes instruction for

verifying that both train related battery chargers are in the same mode (float or equalize) as they both remain attached to their respective vital DC busses as the primary source of DC power to the bus. D is incorrect but plausible. It is conceivable that the Kirk Key interlock arrangement would be in place to protect a 125VDC bus battery charger and the portable battery charger from parallel operation however this is not the case. In fact the operating procedure for transferring a DC bus to it's alternate battery supply includes instruction for verifying that both train related battery chargers are in the same mode (float or equalize), including the portable battery charger, as they would both be attached to their respective vital DC busses as the primary source of DC power to the bus.			
Technical Reference(s):	UFSAR, Section 8.3, Onsite Power Systems, page 77.		OS1048.13, Vital Bus 11A Operation, Section 4.1, Transfering 125 VDC Bus 11A To It's Alternate Battery Supply.
Proposed references to be provided to applicants during examination:			None
K/A Topic:	063 DC Electrical Distribution		
Question Source:	Bank. TEB 20020		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7		
Learning Objective:	L8017I06		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 50	Group #		1		
	K/A #		064 Emergency Diesel Generator K1 Knowledge of the physical connections and/or cause effect relationships between the EDG system and the following systems: K1.02 D/G cooling water system.		
	Importance Rating		3.1		

Proposed Question:

What condition would prevent the Emergency diesel Generator Jacket Water Heat Exchanger Outlet Valve (SW-V-16 or SW-V-18) from opening automatically on an automatic start of its associated Emergency Diesel Generator?

A. Loss of DC power to the valve's solenoid.

B. Loss of air to the valve's solenoid.

C. Emergency Diesel Generator High Speed Relay (375 RPM) fails to energize.

D. Emergency Diesel Generator Low Speed Relay (125 RPM) fails to energize.

Proposed Answer: D

D is correct. The Emergency diesel Generator Jacket Water Heat Exchanger Discharge Valves (SW-V-16 and SW-V-18) auto open when the associated diesel generator speed reaches 125 rpm and the Slow Speed Relay energizes. Failure of the Slow Speed Relay to energize would result in the valve not opening.

A is incorrect but plausible. The valves are solenoid air operated valves however the valves are held closed by a normally energized solenoid. Loss of power to the solenoid would cause the valve to open.

B is incorrect but plausible. The valves are air operated however they are held closed with air pressure. Loss of operating air to the valve would cause it to open.

C is incorrect but plausible. The valve does receive a signal to open based on engine speed however it is the Low Speed Relay vice the High Speed Relay that signals the valve to open.

Technical Reference(s):	1-NHY-301107, sheet E2T/2a, DG Water Jacket HX Valve V16		1-NHY-506831, SW-PAB Cooling Water Train'A' Control Loop

	Schematic Diagram		Diagram

Proposed references to be provided to applicants during examination:	None

K/A Topic:	064 Emergency Diesel Generator
Question Source:	Modified from bank. Teb 30746
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.2 to 41.9/ 45.7 to 45.8
Learning Objective:	L8019I04, L8019I11

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 51	Group #		1		
	K/A #		064 Emergency Diesel Generator A3 Ability to monitor automatic operation of the EDG system, including: A3.07 Load sequencing.		
	Importance Rating		3.6		

Proposed Question:

Given the following plant conditions:

- A loss of offsite power has occurred.
- All equipment has functioned as designed.
- The Emergency Power Sequencer (EPS) is at step 7 when a Safety Injection signal is actuated.

Which of the following describes the EPS response?

A. The LOP loads will be loaded through step 9. The sequencer will then go back to step 0 to sequence the SI loads.

B. All of the loads that were started by the EPS prior to the SI signal will be stripped and reloaded by the EPS along with the SI loads.

C. The SI and LOP loads for steps 8 and 9 will be loaded onto the E-Buses. The EPS will then return to step 0 to load on any unloaded SI loads.

D. When the SI signal is actuated the EPS will immediately return to step 0 to load the SI loads. All LOP loads previously sequenced will remain energized.

Proposed Answer: D

D is correct. If a Safety Injection occurs during the loading sequence all of the previously sequenced loads will remain energized. The sequencer will return to loading step 0 (R2X) and recommence the loading sequence in order to load all LOP and SI loads.

A is incorrect but plausible. It is plausible that the sequencer would be programmed to finish the current sequencing to pick up the remaining LOP loads and then recommence sequencing to pick up any remaining SI loads however this is not the case. The sequencer will immediately return to step 0 as described for answer D.

B is incorrect but plausible. It is plausible that the EPS would strip the EDG loads and recommence

loading the diesel so as to not overload the diesel while picking up SI loads. The sequencer is designed such that the LOP loads can remain energized and not challenge overloading the generator while the SI loads are subsequently sequenced on. C is incorrect but plausible. It is plausible that the sequencer would be programmed to finish the current sequencing to pick up the remaining SI and LOP loads and then recommence sequencing to pick up any remaining SI loads however this is not the case. The sequencer will immediately return to step 0 as described for answer A.			
Technical Reference(s):	1-NHY-310890		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	064 Emergency Diesel Generator		
Question Source:	Bank. Teb 6572		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7/45.5		
Learning Objective:	L8020I21		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 52	Group #		1		
	K/A #		073 Process Radiation Monitoring K3 Knowledge of the effect that a loss or malfunction of the PRM system will have on the following: K3.01 Radioactive effluent releases.		
	Importance Rating		3.6		

Proposed Question:

Given the following plant conditions:

- A release of the 'B' Waste Test Tank is in progress at 45 gallons per minute to the Discharge Transition Structure.
- RDMS Channel R6509(1LM621), Liq Waste Tk to CW Sys fails to the HIGH ALARM condition.

What action will occur?

A. ONLY 1-WL-FCV-1458-1, High Capacity Waste Distillate to Discharge Transition Structure Valve closes.

B. ONLY 1-WL-FCV-1458-2, Low Capacity Waste Distillate to Discharge Transition Structure Valve closes.

C. 1-WL-FCV-1458-1, High Capacity Waste Distillate to Discharge Transition Structure Valve closes and the 'B' Waste Test Tank Pump trips.

D. 1-WL-FCV-1458-2, Low Capacity Waste Distillate to Discharge Transition Structure Valve closes and the 'B' Waste Test Tank Pump trips.

Proposed Answer: A

A is correct. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve. The WTT discharge path has a low capacity valve, 1-WL-FCV-1458-2, which is utilized for discharges where the flow rate is </= 20 gpm. The high capacity valve, 1-WL-FCV-1458-1 is utilized for flow rates > 20 gpm. The stem of the question states that the discharge flow rate is 45 gpm so 1-WL-FCV-1458-1 would be the valve being utilized.

B is incorrect but plausible. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve and would close 1-WL-FCV-1458-2 if it were in service however the stem of the question states that the flow rate is 45 gpm. In this case 1-WL-FCV-1458-1 would be the in service valve.

C is incorrect but plausible. It is true that 1-WL-FCV-1458-1 would close however the high radiation signal does not cause the Waste Test Tank pumps to trip.

D is incorrect but plausible. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve and would close 1-WL-FCV-1458-2 if it were in service however the stem of the question states that the flow rate is 45 gpm. In this case 1-WL-FCV-1458-1 would be the in service valve. Additionally, the high radiation signal does not cause the Waste Test Tank pumps to trip.

Technical Reference(s):	1-NHY-504062, WL-Liquid Effluent Discharge Logic Diagram		1-NHY-504060, WL-Waste Test Tank Pumps Logic Diagram

Proposed references to be provided to applicants during examination:	None
K/A Topic:	073 Process Radiation Monitoring
Question Source:	Bank. TEB 29918
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7/45.6
Learning Objective:	L8086I05

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 53	Group #		1		
	K/A #		076 Service Water K4 Knowledge of the SWS design feature(s) and/or interlock(s) which provide for the following: K4.01 Conditions initiating automatic closure of closed cooling water auxiliary building header supply and return valves.		
	Importance Rating		2.5		

Proposed Question:

Given the following plant conditions:

- The plant was initially at 100% power.
- Both trains of Service Water were aligned to the Cooling Tower per the normal operating procedure.
- Steam Generator "A" completely depressurizes.
- The plant responds as designed.
- The crew has isolated the faulted SG.
- Secondary radiation is normal.
- The crew has verified all AC busses energized from the UATs per the applicable EOP.

What caused SW-V-5, SW Isolation to Secondary Loads, to close?

A. Emergency power sequencer relay PR1.

B. Tower Actuation.

C. Safety Injection Actuation.

D. Reactor Trip P-4 signal.

Proposed Answer: C

C is correct. SW-V-5 is the Train 'B' SW isolation valve for secondary/turbine building loads. The valve will automatically close on either a) Emergency Power Sequencer relay PR1, b) Tower Actuation Signal (low SW system pressure w/ocean SW pump running for >28 seconds) or c) a Safety Injection signal. The stem of the question indicates that a steamline break occurred and that the crew took procedural actions to isolate a faulted steam generator. These conditions indicate that a Safety Injection signal would have actuated.

A is incorrect but plausible. If there is a loss of electrical power and the Emergency Power Sequencer actuates then the PR1 contact would cause SW-V-5 to automatically close however the question stem states that the crew verified emergency bus power from the UAT's so a loss of power and EPS actuation would not have occured.

B is incorrect but plausible. A Tower Actuation signal will cause SW-V-5 to close, however that signal would not have been generated because the SW system was previously aligned to the cooling tower. With the SW system aligned to the cooling tower the ocean SW pump breakers would not be closed so a Cooling Tower actuation signal due to low SW system pressure would not occur. Additionally, the question stem states that the crew verified emergency bus power from the UAT's so a Tower Actuation signal from a loss of power condition would not have occurred.

D is incorrect but plausible. It is plausible that SW to the Turbine Building would isolate automatically on a reactor trip to support heat removal from primary loads as is the case on a Safety Injection signal however a reactor trip P-4 signal does not feed into the automatic close logic for SW-V-5.

Technical Reference(s):	1-NHY-503962, SW-Cooling Tower Actuation Logic Diagram, Sheet 1 of 2		1-NHY-503979, SW-Cooling Tower Actuation Logic Diagram, Sheet 2 of 2
	1-NHY-503977, SW-Isolation Valves For SW Flow to Turbine Building Logic Diagram		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	076 Service Water
Question Source:	Bank. Teb 20216
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.7
Learning Objective:	L8037I06

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 54	Group #		1		
	K/A #		078 Instrument Air K1 Knowledge of the physical connections and/or cause-effect relationships between the IAS and the following systems: K1.02 Service air		
	Importance Rating		2.7		

Proposed Question:

Which of the following describes the expected operation of the Service Air isolation valves, SA-V92 and SA-V93, during an Instrument Air leak?

A. Automatically CLOSE at 80 psig decreasing, automatically REOPEN above 85 psig INCREASING.

B. Automatically CLOSE at 90 psig decreasing, automatically REOPEN above 95 psig INCREASING.

C. Automatically CLOSE at 80 psig decreasing, resets to allow manual OPENING above 85 psig INCREASING.

D. Automatically CLOSE at 90 psig decreasing, resets to allow manual OPENING above 95 psig INCREASING.

Proposed Answer: D

D is correct. SA-V-92/93 will automatically close at <90 psig. The auto close signal will reset at >93 psig which will allow the operator manually reopen the valves at 95 psig.

A is incorrect but plausible. It is true that the valves will automatically close however they do not automatically reopen. Additionally, the associated setpoint is 90 psig versus 80 psig.

B is incorrect but plausible. It is true that SA-V-92/93 will automatically close at <90 psig however the valve will not reopen at >93 psig. The auto close signal will reset at >93 psig allowing manual operation of the valve at 95 psig.

C is incorrect but plausible. It is true that SA-V-92/93 will automatically close and the auto close signal will reset, allowing for manual valve operation however the setpoints are 90/95 psig not 80/85 psig.

Technical Reference(s):	VPRO D5072		1-NHY-503822, SA-Isolation Valves V92 & V93 Logic Diagram

Proposed references to be provided to applicants during examination:			None
K/A Topic:	078 Instrument Air		
Question Source:	Bank. Seabrook 2003 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.2 to 41.9/45.7 to 45.8		
Learning Objective:	L8023I06, L8023I16		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 55	Group #		1		
	K/A #		103 Containment A2 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.03 Phase A and B isolation.		
	Importance Rating		3.5		

Proposed Question:

An event occurs that results in repositioning of multiple components on the main control board. The control room operator, standing at the CVCS (Chemical Volume and Control) portion of the board notes that the following valves have closed:

- CS-V-168, Reactor Coolant Pump Seal Water Return Valve.
- CS-V-150, Letdown Line ORC (Outside Reactor Containment) Isolation Valve.
- CS-V-145, Letdown Regen. Heat Exchanger Isolation Valve.

What event has occurred and what action should be taken?

A. Instrument Air System pressure is degrading. The crew should implement procedure ON1242.01, Loss of Instrument Air.

B. A pressurizer level instrument has failed low. The crew should implement procedure OS1201.07, Pressurizer Level Instrument Failure.

C. An inadvertent Phase A Isolation signal has occurred. The crew should implement procedure OS1205.01, Inadvertent Phase A Containment Isolation.

D. Vital 120VAC Instrument Panel 1A has de-energized. The crew should implement procedure OS1247.01, Loss of a 120VAC Vital Instrument Panel PP-1A, 1B, 1C or 1D.

Proposed Answer: C

C is correct. CS-V-168, Reactor Coolant Pump Seal Water Return Valve is a motor operated valve that automatically closes on a Train 'B' Phase A (T) signal. CS-V-150 is an air operated valve that closes on a Train 'B' Phase A (T) signal. CS-V-145 is an air operated valve that does not automatically close directly from a Phase A (T) signal but is designed to automatically close any time CS-V-150 is not full open. This design feature ensures that the letdown line relief valve does not lift if CS-V-150 closes.

A is incorrect but plausible. If Train B instrument air were degraded then CS-V-150 could close as the valve fails closed on a loss of air. CS-V-145 is an air operated valve that does not automatically close directly from a Phase A (T) signal but is designed to automatically close any time CS-V-150 is not full open. CS-V-168 is a motor operated valve that would not be affected by a loss of instrument air.

B is incorrect but plausible. A failed low pressurizer level channel will cause a letdown isolation. CS-V-145 would close however CS-V-150 would not. The letdown isolation signal is fed to letdown isolation valves RC-LCV 459 and 460 vice the letdown containment isolation valves. Additionally it is conceivable that a low pressurizer level condition would signal the CS-V-168, Reactor Coolant Pump Seal Water Return valve to close as the system is designed to combat loss of inventory from the RCS. CS-V-168 does close on a Safety Injection signal as part of Phase A containment isolation, but does not close based on a low pressurizer level.

D is incorrect but plausible. A loss of vital instrument panel 1A would cause a loss of letdown however it would be based on RC-LCV-459 (Train A) closing vice CS-V-150 closing. Additionally, CS-V-168 is a Train B valve that is not affected by a loss of vital instrument panel 1A as that panel is associated with Train A components.

Technical Reference(s):	OS1205.01, Inadvertant Phase A Containment Isolation		1-NHY-503354, CS-Letdown Line Isol V150 Logic Diagram
	1-NHY-503367, Letdown Isol Valve V145 Logic Diagram		1-NHY-310891, Sheet B72a, RC PP Seal Water Iso Vlv 1-V-168 Schematic Diagram

Proposed references to be provided to applicants during examination:	None
K/A Topic:	103 Containment
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.3/ 45.13
Learning Objective:	L8057I10, L8024I04, L1181I14

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 56	Group #		2		
	K/A #		001 Control Rod Drive K3 Knowledge of the effect that a loss or malfunction of the CRDS will have on the following: K3.02 RCS		
	Importance Rating		3.4		

Proposed Question:

Given the following plant conditions:

- The plant is initially at 100% power.
- Core age is at End of Life.
- Control Rods are in automatic.
- Control Bank 'D' is at 228 steps.
- Subsequently ONE Control Bank 'D' rod drops to the bottom of the core.
- The Reactor does not trip.
- No operator action is taken.

What will be the Reactor Coolant System (RCS) temperature response?

A. RCS temperature will decrease and remain at a new lower value.

B. RCS temperature will initially decrease and then return to program as control rods respond to a Tavg/Rref error signal.

C. RCS temperature will initially decrease and then return to higher than initial value.

D. RCS temperature will initially decrease and then return to program due to positive reactivity from the Moderator Temperature Coefficient.

Proposed Answer: A

A is correct. Control Bank D is at 228 steps which is above the C-11 setpoint of 223 steps. C-11 blocks automatic control rod withdrawal above 223 steps on Bank 'D'. When the control rod drops reactor coolant temperature will decrease due to the addition of negative reactivity. With the core at End of Life there is a negative moderator temperature coefficient. The decrease in RCS temperature would add positive reactivity due to the MTC but not enough to return reactor power/RCS temperature back to program.

B is incorrect but plausible. If control rods were capable of automatic withdrawal then the lowering RCS temperature and disparity between nuclear power and Tref power would call for outward rod motion. Control Bank D is at 228 steps which is above the C-11 setpoint of 223 steps. C-11 blocks automatic control rod withdrawal above 223 steps on Bank 'D'.

C is incorrect but plausible. If control rods were capable of automatic withdrawal then the lowering RCS temperature and disparity between nuclear power and Tref power would call for outward rod motion. Control Bank D is at 228 steps which is above the C-11 setpoint of 223 steps. C-11 blocks automatic control rod withdrawal above 223 steps on Bank 'D'. Temperature would not return to a value higher than original.

D is incorrect but plausible. It is true that RCS temperature would decrease however the effects of the moderator temperature coefficient would add positive reactivity at the new lower temperature. Temperature would not return to program.

Technical Reference(s):	Primary Technical Data Book, RE-6, Moderator Temperature Coefficient vs. Burnup		1-NHY-509049, Rod Control and Blocks w/ Functional Diagram

Proposed references to be provided to applicants during examination:	None

K/A Topic:	001 Control Rod Drive

Question Source:	Modified from bank. Teb 26912			

Question Cognitive Level:	Higher: Comprehension/Analysis		

10 CFR Part 55 Content:	41.7/45.6	

Learning Objective:	L8031I22

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 57	Group #		2		
	K/A #		011 Pressurizer Level Control A3 Ability to monitor automatic operation of the PZR LCS, including: A3.03 Charging and Letdown		
	Importance Rating		3.2		

Proposed Question:

Due to an air leak instrument air pressure is lost to CS-V-145, Letdown Regenerative Heat Exchanger Isolation Valve. How will the charging system respond, if at all?

A. Air leak has no effect on the system.

B. CS-FCV-121 will throttle CLOSED to decrease charging flow and RCP seal injection flow will DECREASE.

C. CS-FCV-121 will throttle CLOSED to decrease charging flow and RCP seal injection flow will INCREASE.

D. CS-FCV-121 will throttle OPEN to increase charging flow and RCP seal injection flow will INCREASE.

Proposed Answer: B

B is correct. If letdown flow is isolated then pressurizer level will increase. As pressurizer level increases CS-FCV-121 will throttle closed to reduce charging flow into the reactor coolant system. The seal injection branch line is downstream of CS-FCV-121. Any decrease in charging flow will also result in a decrease in seal injection flow.

A is incorrect but plausible. The candidate may identify that CS-V-145 is a letdown component and not directly associated with the charging system or its controls. The system would be effected as a loss of letdown would cause a pressurizer level increase resulting in a response by the level control system.

C is incorrect but plausible. It is true that as pressurizer level increases CS-FCV-121 will throttle closed to reduce charging flow into the reactor coolant system. The seal injection branch line is downstream of CS-FCV-121 vice upstream so seal injection flow would decrease. The physical layout of the charging/seal injection piping and interaction/interface of CS-FCV-121 and seal injection flow (including the operation of seal injection hand control valve CS-HCV-181) is a common operator misconception at Seabrook Station.

D is incorrect but plausible. It is true that if CS-FCV-121 throttled open then seal injection flow would increase however CS-FCV-121 would throttle closed if letdown flow isolated.						
Technical Reference(s):	1-NHY-509027, Pressurizer Level Control Process Control Block Diagram					PID-1-CS-B20722, Chemical and Volume Control Sys Heat Exchangers Detail
	PID-1-CS-B20725, Chemical and Volume Control Charging System Detail					
Proposed references to be provided to applicants during examination:						None
K/A Topic:	011 Pressurizer Level Control					
Question Source:	Bank.Teb 20753					
Question Cognitive Level:	Higher: Comprehension/Analysis					
10 CFR Part 55 Content:	41.7/45.5					
Learning Objective:	L8027I06					

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 58	Group #		2		
	K/A #		014 Rod Position Indication A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: A1.02 Control rod position indication on control room panels.		
	Importance Rating		3.2		

Proposed Question:

The control room operator is in the process of manually inserting control rods when the following Digital Rod Position Indication (DRPI) system indications appear:

- The General Warning LED light for rod H8 is FLASHING.
- The Urgent Alarm LED's are FLASHING.
- The Rod Bottom light for rod H8 is LIT.
- The Rod Deviation LED's are LIT.

What condition has caused these indications?

A. Rod H8 has dropped to the bottom of the core.

B. There is a Data A and Data B failure for rod H8 DRPI indication.

C. Rod H8 is misaligned from the rest of the rods in it's bank by more than 12 steps.

D. Rod H8 is misaligned from the rest of the rods in it's group by more than 38 steps.

Proposed Answer: B

B is correct. Loss of DRPI data capability will cause a General Warning indication for the affected rod. Additionally, if Data A and Data B are failed then there will be Rod Deviation, Urgent Alarm and Rod Bottom (for the affected rod) indications.

A is incorrect but plausible. It is true that a dropped rod will cause Rod Deviation and Dropped Rod (for the affected rod) indications. Urgent Alarm indication and General Warning indication are indicative of DRPI system data failure vice an actual dropped or misaligned rod.

C is incorrect but plausible. It is true that a rod misaligned by more than 12 steps from it's bank will cause Rod deviation indication. A misaligned control rod will not cause a Rod Bottom indication. Rod Bottom indication is only received for an actual dropped rod or in the case of DRPI Data A and Data B failure. Urgent Alarm indication and General Warning indication are indicative of DRPI system data failure vice an actual dropped or misaligned rod.

D is incorrect but plausible. It is true that alarm conditions will be received if the sum of Data A and Data B is greater than 38, however this is associated with an urgent failure data condition vice an actual 38 step rod misalignment.

Technical Reference(s):	OS1210.07, RPI Malfunction		
Proposed references to be provided to applicants during examination:	None		
K/A Topic:	014 Rod Position Indication		
Question Source:	New		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.5/45.5		
Learning Objective:	L8032I08		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 59	Group #		2		
	K/A #		015 Nuclear Instrumentation 2.4.4 Ability to recognize abnormal indications from system operating parameters that are entry level conditions for emergency and abnormal operating procedures.		
	Importance Rating		4.5		

Proposed Question:

Given the following plant conditions:

- The plant is in Mode 2.
- Reactor power is stable at 4%.
- Power Range Channel NI-41 is undergoing maintenance with the control power fuses removed.
- Power Range Channel NI-43 fails high.

What is the affect on source range instrumentation and what operator action is required?

A. Both Source Range Instruments will remain de-energized. Enter E-0, “Reactor Trip or Safety Injection”.

B. Both Source Range Instruments remain energized. Enter E-0, “Reactor Trip or Safety Injection”.

C. Both Source Range Instruments will remain de-energized. Within 1 hour determine by observation that the Power Above P-10 Block Trips annunciator is in its required state.

D. Both Source Range Instruments remain energized. Within 1 hour determine by observation that the Power Above P-10 Block Trips annunciator is in its required state.

Proposed Answer: A

A is correct. Power range channel NI-41 has its control power fuses removed which puts it in a tripped condition. When power range channel NI-43 fails high there is now a 2 of 4 logic met for the Power Range High Flux Low Setpoint (25% power)Trip. Additionally, the 2 of 4 channel coincidence for signal P-10 (10% power) is met. The source range instruments will remain de-energized. With the reactor trip signal actuated procedure E-0, Reactor Trip or Safety Injection should be implemented.

B is incorrect but plausible. When power range channel NI-43 fails high there is now a 2 of 4 logic met for the Power Range High Flux Low Setpoint (25% power). With the reactor trip signal actuated procedure E-0, Reactor Trip or Safety Injection should be implemented. This answer is wrong however as the source range instruments would be automatically deenergized by P-10. There is a common operator misconception regarding the interface between P-10 and the source range instruments. P-10 automatically deenergizes the source range instruments vice allows for manual deenergization, as would be the case with the P-6 interlock.

C is incorrect but plausible. The 2 of 4 channel coincidence for signal P-10 (10% power) is met. The source range instruments will remain de-energized. The 1 hour requirement to verify proper P-10 status is plausible as the instrument failure in the stem of the question would cause a violation of the tech spec required "Minimum Channels Operable" for the P-10 function however that Limiting Condition For Operation is only applicable in Mode 1. Given the conditions in the stem of the question the reactor will trip and the plant would no longer be in Mode 1.

D is incorrect but plausible. If the reactor remained at power the 1 hour requirement to verify proper P-10 status would be a plausible action as the instrument failure in the stem of the question would cause a violation of the tech spec required "Minimum Channels Operable" for the P-10 function. The answer is wrong however because when power range channel NI-43 fails high there is then a 2 of 4 logic met for the Power Range High Flux Low Setpoint (25% power). With the reactor trip signal actuated procedure E-0, Reactor Trip or Safety Injection should be implemented.

Technical Reference(s):	E-0, Reactor Trip or Safety Injection, Part B, Symptoms or Entry Conditions		E-0, Reactor Trip or Safety Injection, Attachment B, Symptoms that Require a Reactor Trip.
	1-NHY-509043, NI and Manual Trip Signals w/ Functional Diagram		1-NHY-509044, NI Perm. And Blocks w/ Functional Diagram

Proposed references to be provided to applicants during examination:	None
K/A Topic:	015 Nuclear Instrumentation
Question Source:	Modified from bank. Teb 31057
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.2/45.6
Learning Objective:	L1202I01, L8030I08, L8030I09, L8030I12

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 60	Group #		2		
	K/A #		029 Containment Purge A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the Containment Purge System controls including: A1.03 Containment pressure, temperature, and humidity.		
	Importance Rating		3.0		

Proposed Question:

The crew is implementing procedure FR-Z.3, "Response to High Containment Radiation Level." What condition is used to determine if the Containment Recirculation Filter should be placed in service and why?

A. Containment pressure is verified less than 18 psig because a 'P' signal will prevent recirculation filter realignment to the Filter Mode.

B. Containment pressure is verified less than 4.3 psig because the recirculation filter is not designed to be used in an adverse containment atmosphere.

C. Containment hydrogen concentration is verified less than 4% to prevent the possibility of hydrogen gas ignition inside the recirculation/filter system.

D. Containment radiation level is verified less than the Containment Radiation Monitor setpoint to minimize the potential for a radiological release from containment.

Proposed Answer: A

KA match justification: NUREG-1122, 029 Containment Purge, KA Section K2 includes "recirculation fans" and KA section K1 includes the "recirculation system". This indicates that the recirc/filtration subsystem is considered part of KA category 029 Containment Purge. Additionally, Seabrook Station includes the containment recirculation/filtration components and the containment air purge components as subsystems of a common Containment Air Handling System.

A is correct. A 'P' signal will prevent operation of the equipment in the Filter Mode.

B is incorrect but plausible. It is true that the filter is not designed to be used at elevated pressure, however the setpoint is 18 psig ('P' signal).

C is incorrect but plausible. Containment hydrogen concentration is a concern during LOCA conditions, however it is associated with operation of the hydrogen recombiners vice the recirculation filter. D is incorrect but plausible. Containment radiation level is the primary concern, however it is addressed by verifying the containment purge valve isolation vice operation of the recirculation filter.

Technical Reference(s):	1-NHY-503204, CAH-Containment Structure Recirc Filter Fan Logic Diagram		FR-Z.3, Response to High Containment Radiation Level

Proposed references to be provided to applicants during examination:	None
K/A Topic:	029 Containment Purge
Question Source:	Modified from bank. Teb 26697
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.5/45.5
Learning Objective:	L8038I04

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 61	Group #		2		
	K/A #		041 Steam Dump/Turbine Bypass Control K4 Knowledge of the SDS design feature(s) and/or interlock(s) which provide for the following: K4.09 Relationship of low/low Tavg setpoint in SDS to primary cooldown.		
	Importance Rating		3.0		

Proposed Question:

Given the following plant conditions:

- A plant cooldown is in progress.
- Reactor Coolant System loop Tavg:
  - Loop 1: 550°F and decreasing
  - Loop 2: 548°F and decreasing
  - Loop 3: 551°F and decreasing
  - Loop 4: 548°F and decreasing
- Steam header pressure is 1030 psig and decreasing.
- Steam Dump Mode Selector Switch is in Steam Pressure Mode.
- Steam Dump Controller is in Manual set at 30% demand.
- The operator momentarily places the Train 'A' and Train 'B' Steam Dump Bypass Interlock Switches to 'Bypass' and then releases them.

What is the status of the Steam Dump valves following the operator's actions?

A. All valves are fully closed.
B. Three valves in Group 1 are partially open.
C. Three valves in Group 1 are fully open and the valves in Group 2 are fully closed.
D. Three valves in Group 1 are fully open and three valves in Group 2 are partially open.

Proposed Answer: C

C is correct. The P-12 Low-Low Setpoint is at 550°F. At this point taking the Steam Dumps to Bypass Interlock allows for operation of the Group 1 valves only. At 30% output on the controller the Group 1 valves would be full open.

A is incorrect but plausible. The RCS loop temperatures listed in the stem of the question indicate that the P-12 setpoint/coincidence has been met. This would cause all the steam dump valves to close. If the student had a misconception regarding the operation of the Bypass Interlock switch operation they may believe that all of the valves would remain closed. B is incorrect but plausible. It is true that the Group 1 valves would function however, with a 30% demand signal the Group 1 valves would be full open as then throttle through a 0-25% demand signal range. D is incorrect but plausible. It is true that the Group 1 valves would be fully open however the Group 2 valves are not designed to function below P-12 even with Bypass Interlock actuated.			
Technical Reference(s):	1-NHY-509050, MS Dump Control w/ Functional Diagrams		
Proposed references to be provided to applicants during examination:		None	
K/A Topic:	041 Steam Dump/Turbine Bypass Control		
Question Source:	Modified from Bank. Byron 2006 NRC Exam		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.7		
Learning Objective:	L8047I12, L8047I05, L8047I06		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 62	Group #		2		
	K/A #		045 Main Turbine Generator K5 Knowledge of the operational implications of the following concepts as they apply to the MT/G System: K5.17 Relationship between MTC and boron concentration in RCS as turbine load increases.		
	Importance Rating		2.5		

Proposed Question:

Given the following plant conditions:

- The plant is at 90% power.
- Control rods are at 228 steps.
- Boron concentration is 1600 ppm.

A control valve malfunction results in a step 50 MWe increase in load.

How would the initial plant response be different if the same 50 MWe step increase occurred when boron concentration was 300 ppm?

A. The RCS temperature change would be smaller.

B. More positive reactivity would be added.

C. The power change would be smaller.

D. The power defect would be smaller.

Proposed Answer: A

A is correct. As load is increased, RCS temperature will decrease. The temperature decrease will add positive reactivity. At BOL a small moderator temperature coefficient will add less reactivity per degree change than at EOL. The temperature change at EOL will be less (more reactivity added per °F).

B is incorrect but plausible. The power change is the same so the amount of positive reactivity that must be added by MTC is the same. At EOL the MTC is larger (more negative) so there would be less temperature change at EOL.

C is incorrect but plausible. The power change is the same. 50 MWe is approximately a 3.8% change (based on 1295 MWe). This remains the same at EOL. D is incorrect but plausible. At EOL the power defect from a 3.8% load change is approximately 110 pcm but only approximately 58 pcm at BOL.			
Technical Reference(s):	RE-6, Moderator Temperature Coefficient vs. Burnup		RE-8, Total Power Defect vs. Power and Boron Concentration
Proposed references to be provided to applicants during examination:			None
K/A Topic:	045 Main Turbine Generator		
Question Source:	Bank. Diablo Canyon 2007 NRC Exam		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.5/45.7		
Learning Objective:	L1402I01, L1402I05, L1402I06		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 63	Group #		2		
	K/A #		055 Condenser Air Removal K1 Knowledge of the physical connections and/or cause effect relationship between the CARS and the following: K1.06 PRM system		
	Importance Rating		2.6		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- RM-6505, Condenser Air Evacuation Discharge Radiation Monitor is in ALARM.

Which of the following describes the significance of this alarm?

A. Radiation level on RM-6505 isolates SG Blowdown .

B. Radiation level on RM-6505 provides an input to the calculation for an approximate value of Primary to Secondary Leak Rate.

C. Radiation level on RM-6505 is used to determine the need for a reactor trip and SI per OS1227.02, 'Steam Generator Tube Leak'.

D. Radiation level on RM-6505 is used to determine which secondary systems need to be isolated per OS1227.02, 'Steam Generator Tube Leak'.

Proposed Answer: B

B is correct. RM-6505 is used for the calculated primary to secondary leak rate and also used in OS1227.02 if the value must be calculated manually.

A is incorrect but plausible. There are radiation monitors associated with SG Blowdown that will isolate the systems flash tank however RM-6505 does not provide that function.

C is incorrect but plausible. The leak rate is used to determine the rate of plant downpower however reactor trip and SI criteria are based on the threshold of maintaining >7% pressurizer level utilizing two charging pumps.

D is incorrect but plausible. RM-6505 indications are indicative of primary to secondary leakage, however it is indicative of leakage into the steam generators and not indicative of any specific secondary system.

Technical Reference(s):	OS1227.02, Steam Generator		

	Tube Leak		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	055 Condenser Air Removal		
Question Source:	Bank. Seabrook 2009 Remediation Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	CFR 41.2 to 41.9/45.7 to 45.8		
Learning Objective:	L1190I02		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 64	Group #		2		
	K/A #		056 Condensate A2 Ability to (a) predict the impacts of the following malfunctions or operations on the condensate system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: A2.04 Loss of condensate pumps.		
	Importance Rating		2.5		

Proposed Question:

Given the following plant conditions:

- The plant is initially at 75% power.
- The Unit Auxiliary Transformer feeder breaker to Bus 3 opened and the Reserve Auxiliary Transformer feeder breaker failed to close.
- Control Rods are inserting in automatic.
- T_{avg} is approximately 584°F.
- Condenser Steam Dumps are open.
- Steam Generator pressures are approximately 1100 psig and increasing.
- Alarm point D7761, CTL ROD BANK D INSERTION LIMIT LOW is in alarm.
- Alarm point D4421, TAVG-TREF DEVIATION is in alarm.

What procedure should be utilized to address these plant conditions?

A. OS1202.04, 'Rapid Boration.'

B. OS1231.03, 'Turbine Runback/Setback.'

C. OS1233.01, 'Loss of Condenser Vacuum.'

D. OS1290.02, 'Response to Secondary System Transient.'

Proposed Answer: B

B is correct. Condensate Pumps 30A and 30C are powered from Bus 3. A Turbine Setback to 55% power is initiated based on 2 of 3 Condensate Pump breakers open.

A is incorrect but plausible. A rapid boration is required if control rods insert to below the Rod

Insertion Limit, however this condition is indicated by the Rod Insertion Limit Low-Low alarm vice the Low alarm.

C is incorrect but plausible. A loss of Circulating Water Pumps could cause a loss of condenser vacuum. 2 of the 3 CW pumps are supplied power from UAT 2A and RAT 3A however they are powered from Bus 1 and are supplied via separate UAT/RAT feeder breakers than those of Bus 3.

D is incorrect but plausible. OS1290.02, Response to Secondary Plant Transient does provide guidance for transient conditions within the secondary plant, however the Loss of 2 of 3 Condensate Pump runback signal is a specific entry condition for OS1231.03, 'Turbine Runback/Setback'. Additionally, OS1231.03 contains high level actions to address the question stem conditions, including rod control response, steam dump operation, steam generator pressures and rod insertion limit.

The stem of the question includes conditions indicative of a plant setback due to loss of 2/3 condensate pumps. Additionally, the question stem states that a) control rods are inserting, b) steam dumps are open and that the rod insertion limit low alarm has actuated. The entry conditions for OS1231.03 include a) Steam dump arming on C-7 and b) UL status lamp lit for a turbine runback/setback condition. The question stem supports these procedure entry conditions. Furthermore, OS1231.03 includes procedural steps/strategies for addressing a) proper rod control response, b) proper steam dump operation, c) steam generator pressures,d) response to a possible rod insertion limit LO-LO alarm and e) identifying the cause of the turbine runback/setback. These strategies all address plant conditions listed in the question stem.

Procedure OS1290.02, 'Response to Secondary System Transient includes the following entry conditions:

- Abnormal feedwater or condensate heater level oscillations.
- Abnormal secondary system flow transients.
- Automatic start of the standby condensate pump.
- Automatic isolation of a condensate or feedwater heater.
- Computer related alarms for secondary system transients affecting condensate or feedwater heaters.
- Heater drain pump seal failures.

The conditions in the question stem do not specifically match any of these entry conditions. Furthermore, the steps/strategies within the body of the procedure are structured to address flown and level transients within the condensate and feedwater heater strings and do not specifically address a) proper rod control response, b) proper steam dump operation, c) steam generator pressures,d) response to a possible rod insertion limit LO-LO alarm or e) identifying the cause of the turbine runback/setback.

Technical Reference(s):	OS1231.03, Turbine Runback/Setback		MPCS Graphic, Electrical Distribution Overview.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	056 Condensate			
Question Source:	Bank. Teb 23090			
Question Cognitive Level:	Higher: Comprehension/Analysis			
10 CFR Part 55 Content:	41.5/43.5/45.3/45.13			
Learning Objective:	L1183I09			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		2		
Question 65	Group #		2		
	K/A #		075 Circulating Water A2 Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: A2.02 Loss of circulating water pumps.		
	Importance Rating		2.5		

Proposed Question:

Given the following plant conditions:

- The plant is at 50% power.
- CW-P-39A and CW-P-39B, Circulating Water Pumps are in service.
- Subsequently the control room receives the following alarms:
  - D4000, CW Screen A Fouled
  - D4001, CW Screen B Fouled
- 1 minute later CW-P-39A trips.
- The crew enters procedure ON1238.01, “Circulating Water System Malfunction.”

What action should be taken next?

A. Transfer the Service Water System to the Cooling Tower.

B. Trip the reactor and go to E-0, “Reactor Trip or Safety Injection”.

C. Reduce plant power as necessary to comply with Circulating Water ΔT limits.

D. Reduce plant power as necessary to maintain Main Condenser vacuum greater than 25 inches.

Proposed Answer: B

B is correct. Per ON1238.01, step 6, if less than 2 CW pumps are running then the reactor should be tripped and procedure E-0, Reactor Trip or Safety Injection should be implemented. This step supports the precaution statement listed in procedure ON1038.01, Circulating Water System Pump

Startup which states “Operation of one CW pump with more than one waterbox in service should be minimized to prevent pump damage due to pump runout.

A is incorrect but plausible. If the CW pump configuration were not down to 1 pump then the crew would not need to trip the reactor and procedural steps for mitigating the fouled CW screens would be appropriate. These procedural steps include transferring SW to the Cooling Tower.

C is incorrect but plausible. If there were 3 CW pumps in operation and 1 tripped then there are procedural steps for checking CW system conditions. These steps include the guidance for reducing plant power as necessary to comply with CW ΔT limitations.

D is incorrect but plausible. If the plant were to stay on line then the crew could implement procedural steps for mitigating fouled screen conditions. The procedural flowpath in this instance includes the major action step for reducing plant load as necessary to maintain condenser vacuum greater than 25 inches.

Technical Reference(s):	ON1238.01, Circulating Water System Malfunction.		ON1038.01, Circulating Water System Pump Startup, Precuation 3.6.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	075 Circulating Water

Question Source:	New			

Question Cognitive Level:	Higher: Comprehension/Analysis		

10 CFR Part 55 Content:	41.5/43.5/45.3/ 45.13	

Learning Objective:	L1188I14, L8053I12

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 66	Group #		1		
	K/A #		2.1.1 Knowledge of conduct of operations requirements.		
	Importance Rating		3.8		

Proposed Question:

In accordance with OP 9.2, Transient Response Procedure User's Guide, which one of the following is considered a "Skill of the Operator Task"?

A. Placing Rod Control in Manual or Automatic.

B. Re-opening Letdown Isolation Valves in response to an instrument failure.

C. Performing a rapid boration in response to excessive Control Rod insertion.

D. Swapping input channels to a Main Feedwater Regulating Valve controller in response to an instrument failure.

Proposed Answer: A

A is correct. Per OP 9.2, Transient Response Procedure User's Guide, section 4.9.5 placing rod control in manual or automatic is listed on the "complete list" of skill of the operator tasks.

B is incorrect but plausible. Operation of Letdown components are listed, however it is to support isolating letdown via CS-V-145 or adjusting charging or letdown flows. There is no allowance for realigning letdown valves that have closed due to an instrument failure. Specific instructions for that process are delineated in the Pressurizer Level Instrument Failure abnormal procedure.

C is incorrect but plausible. There is guidance for placing rod control in manual, however there is no guidance for initiating a rapid boration in response to excessive Control Rod insertion. If the Rod Insertion Low Low alarm were to actuate the Alarm Response Procedure contains the required guidance for initiating a rapid boration to mitigate a loss of shutdown margin condition.

D is incorrect but plausible. There is guidance for taking manual control of any component using the manual/automatic controller station, as would be prudent in the event of a failed instrument input into the Main Feedwater Reg. Valves, however the specific guidance for swapping input channels is directed by the applicable abnormal operating procedure.

Technical Reference(s):	OP 9.2, Transient Response Procedure User's Guide, Section 4.9.5		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	2.1.1 Knowledge of conduct of operations requirements.				
Question Source:	Modified from bank. Seabrook 2007 NRC Exam.				
Question Cognitive Level:	Memory or Fundamental Knowledge				
10 CFR Part 55 Content:	41.10/45.13				
Learning Objective:	L1505I23				

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 67	Group #		1		
	K/A #		2.1.28 Knowledge of the purpose and function of major system components and controls.		
	Importance Rating		4.1		

Proposed Question:

OS1000.10, "Operation At Power", Figure 16, "Forcing Pressurizer Sprays" contains a note that states the following:

> "Pressurizer spray flow should be initiated any time RCS boron concentration changes by greater than 50 ppm."

What is the reason for this statement?

A. Prevents boron accumulation in the pressurizer spray nozzles.

B. Prevents an unintended reactivity change from occurring during a pressurizer outsurge.

C. Ensures that the boron concentration used in the accident analyses remains within the analyzed range of values.

D. Ensures proper mixing occurs such that RCS loop boron samples accurately reflect actual boron concentration.

Proposed Answer: B

B is correct. Pressurizer heater operation causes continuous PZR spray which ensures a minimum boron concentration differential between the PZR and the RCS loops precluding an inadvertent boron dilution event as a result of a PZR outsurge of water into the RCS.

A is incorrect but plausible. This phenomenon is plausible as initiating sprays could prevent buildup of boron on spray nozzles and create a possible future reactivity concern or concern with nozzle function.

C is incorrect but plausible. This statement reflects the boron concentration of the entire RCS not just the PZR and initial boron concentration of the RCS is not considered in the UFSAR accident analyses.

D is incorrect but plausible. PZR spray operation does ensure proper mixing of boron however it is within the PZR and the RCS to prevent a dilution event in the case of a pressurizer outsurge.

Technical Reference(s):	OS1000.01, Heatup From Cold Shutdown to Hot Standby.		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	2.1.28 Knowledge of the purpose and function of major system components and controls.		
Question Source:	Bank. Teb 24482		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.7		
Learning Objective:	L1167I05		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 68	Group #		1		
	K/A #		2.1.37 Knowledge of procedures, guidelines or limitations associated with reactivity management.		
	Importance Rating		4.3		

Proposed Question:

Per procedure OS1000.07, "Approach to Criticality" what reactivity control parameters are verified prior to each 50 step incremental control rod withdrawal?

A. RCS boron is verified to be within 15 ppm of critical boron. All Shutdown Rods are verified to be fully withdrawn.
B. RCS boron is verified to be within 15 ppm of critical boron. Critical Rod Height is predicted to be above the Rod Insertion Limit.
C. All Shutdown Rods are verified to be fully withdrawn. The lowest operating loop T_{avg} is verified to be greater than 551°F.
D. The lowest operating loop T_{avg} is verified to be greater than 551°F. Critical Rod Height is predicted to be above the Rod Insertion Limit.

Proposed Answer: C

C is correct. Per OS1000.07, Approach to Criticality, step 4.4.5, all shutdown rods are verified withdrawn within 15 minutes of any control bank withdrawal. Additionally, the lowest operating loop T_{avg} is verified to be greater than 551°F every 15 minutes until the reactor is declared critical. OS1000.07 includes specific steps to perform these verifications prior to each 50 step incremental control rod withdrawal. These verifications are pursuant to technical specification reactivity limitations associated with a) ensuring adequate Shutdown Margin and b) ensuring that moderator temperature coefficient is within it's analyzed temperature range.

A is incorrect but plausible. It is true that all shutdown rods are verified withdrawn within 15 minutes of any control bank withdrawal. It is also true that procedure OS1000.07 directs verifying RCS boron concentration within 15 ppm of critical boron, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate.

B is incorrect but plausible. It is true that procedure OS1000.07 directs verifying RCS boron concentration within 15 ppm of critical boron, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate. It is also true that procedure OS1000.07 directs verifying that the estimated critical rod height is predicted to be above the rod insertion limit, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical

Position data is accurate. D is incorrect but plausible. It is true that the lowest operating loop T_{avg} is verified to be greater than 551°F prior to each incremental control rod withdrawal. It is also true that procedure OS1000.07 directs verifying that the estimated critical rod height is predicted to be above the rod insertion limit, however this is required to be done within 4 hours of performing an approach to criticality as part of verifying that the Estimated Critical Position data is accurate.			
Technical Reference(s):	OS1000.07, Approach to Criticality		
Proposed references to be provided to applicants during examination:		None	
K/A Topic:	2.1.37 Knowledge of procedures, guidelines or limitations associated with reactivity management.		
Question Source:	New		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.1/43.6/45.7		
Learning Objective:	L1162I02		

Seabrook Station 2010 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 69	Group #		2		
	K/A #		2.2.12 Knowledge of surveillance procedures.		
	Importance Rating		3.7		

Proposed Question:

According to surveillance procedure OX1401.02, "RCS Steady State Leak Rate Calculation", which of the following parameters is an input into the calculation for IDENTIFIED RCS leakage?

A. Pressurizer Level
B. Integrated Makeup Total.
C. Containment Sump Level.
D. Pressurizer Relief Tank Level.

Proposed Answer: D

D is correct. The surveillance specifies the difference between the two variables in order to comply with Technical Specifications. According to Form B, the PRT is a source of identified RCS leakage.

This procedure is used by operations every 12 hours to determine sources of RCS leakage. Common misconceptions surrounding sources of IDENTIFIED versus UNIDENTIFIED leakage have in the past led to inaccuracies in these calculations.

All distractors are incorrect but plausible since they are referenced in OX1401.02 as sources of UNIDENTIFIED RCS leakage. The candidate must be able to distinguish between the two sources of which all distractors are credible unidentified sources.

Technical Reference(s):	OX1401.02, RCS Steady State Leak Rate Calculation		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	2.2.12 Knowledge of surveillance procedures.
Question Source:	Bank. Seabrook 2005 NRC Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.10/43.5/45.13
Learning Objective:	L8010I10

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 70	Group #		2		
	K/A #		2.2.43 Knowledge of the process used to track inoperable alarms.		
	Importance Rating		3.0		

Proposed Question:

Which of the following is <u>NOT</u> an allowed administrative method for removing a computer point from service per ON1090.06, "Use and Control of Deleted Analog and Digital Points?"

A. Deleting from scan computer point D7325, 'Bus 17 Voltage Low' per an authorized clearance order.

B. Deleting from scan computer point A2858, 'Incore Thermocouple H-03' and tracking it in the Component Deviation Log because its sensor is disconnected.

C. Deleting from alarm computer point F8212, 'SW Cathodic Protection System Trouble' per an active work order because the alarm was continuously cycling.

D. Deleting from alarm computer point D5181, 'Spent Fuel Pool Skimmer Pump Flow Low' per OS1014.02, "Operation of Spent Fuel Pool Cooling and Purification System."

Proposed Answer: B

B is correct. Per procedure ON1090.06, Use and Control of Deleted Analog and Digital Points computer points can be removed from service by one of the following methods: a) per procedure, b) per an authorized TMOD, c) per a Work Order, d) per a Clearance Order or e) if not by any of the above methods then with a completed 50.59 Applicability Determination.

All of the distractors are plausible as there has been a historic operator misconception with regard to proper tracking/configuration control regarding removal of main plant VAS computer points from service. A, B, and C are all incorrect as they include allowed methods of removing alarm points from service.

Technical Reference(s):	ON1090.06, Use and Control of Deleted Analog and Digital Points-Figure 1		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	2.2.43 Knowledge of the process used to track inoperable alarms.
Question Source:	New
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55	41.10/43.5/45.1

Content:	3	
Learning Objective:		

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

Proposed Question:

Given the following plant conditions:

- Containment building pressure is being reduced in accordance with OS1023.69, "Containment On-Line Purge (COP) System Operation."
- COP Exhaust Containment Isolation Valves COP-V-3 and COP-V-4 have been opened.
- The crew is establishing COP flow through COP-V-8, COP Exhaust Throttle Valve (Coarse Control).
- RM-6527A-1 and RM-6527A-2, Train "A" COP go into HIGH alarm.
- ALL systems function as designed.

Which of the following describes how the control room crew will control the radiological release?

A. Control room operators must ensure COP-V-4 automatically closes to stop the release.

B. Control room operators must ensure COP-V-3 and COP-V-8 automatically close to stop the release.

C. Control room operators must ensure COP-V-4 and COP-V-8 automatically close to stop the release.

D. Control room operators must manually close COP-V-3 and COP-V-4 since no automatic actions will occur.

Proposed Answer: A

A is correct. COP-V-3 and 4 receive an automatic CVI signal to close when high radiation is sensed. COP Valves 1 & 4 receive a CVI signal from Train 'A'. COP Valves 2 & 3 receive a CVI signal from Train 'B'.

B is incorrect but plausible. COP V-3 does receive a CVI signal, however it is from Train B radiation monitors. It is a common operator misconception that the COP exhaust throttle valves also receive a CVI signal but this is incorrect (COP-V-8 will not close automatically).
C is incorrect but plausible. COP-V-4 is a Train "A"valve and will receive a CVI signal to close. It is a common operator misconception that the COP exhaust throttle valves also receive a CVI signal but this is incorrect (COP-V-8 will not close automatically).
D is incorrect but plausible. Both COP-V- 3 and 4 receive a CVI signal to close, however COP-V-3 receives it's signal from Train 'B'.

Technical Reference(s):	1-NHY-503298, COP-Containment Isolation Valves Logic Diagram		1-NHY-310920, Sheet BSOa, Containment Purge Exhaust Valve 1-V-8 Schematic Diagram
1-NHY-509048, Safeguard Actuation Signals w/ Functional Diagrams	1-NHY-310920, Sheet E87, Containment Online Purge System A Train Vital Control Schematic Diagram		1-NHY-310920, Sheet E88, Containment Online Purge System B Train Vital Control Schematic Diagram
Proposed references to be provided to applicants during examination:			None
K/A Topic:	2.3.11 Ability to control radiation releases.		
Question Source:	Bank. Seabrook 2005 NRC Exam		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.11/43.4/45.10		
Learning Objective:	L8059I06		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 72	Group #		3		
	K/A #		2.3.13 Knowledge of the radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.		
	Importance Rating		3.4		

Proposed Question:

An accessible area of the Primary Auxiliary Building has a general area dose rate of 500 mrem/hr. What posting should be displayed at the entrance to this area?

A. Radiation Area

B. High Radiation Area

C. Very High Radiation Area

D. Tech. Spec. Locked High Radiation Area

Proposed Answer: B

B is correct per SSRP, Chapter 1, Section 3.4.6, Area Designations.

A,C, and D are all incorrect but plausible. The following are the criteria for area designations:

Radiation Area-Any area accessible to individuals in which radiation levels could result in the individual receiving a dose in excess of 5mrem (DDE) in one-hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

High Radiation Area- Any area accessible to individuals in which radiation levels could result in the individual receiving a dose in excess of 100mrem (DDE) in one-hour at 30 cm from the radiation source or from any surface that the radiation penetrates.

Very High Radiation Area-A high radiation area accessible to individuals, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one-hour at 1 meter from the radiation source or from any surface that the radiation penetrates.

Tech. Spec. Locked High Radiation Area- Any high radiation area (1) accessible to individuals in which radiation levels could result in an individual receiving a dose equivalent of >1000mrem

(DDE) in one-hour at 30 cm from the radiation source or from any surface that radiation penetrates and (2) not meeting the requirements of a Very High Radiation Area.			
Technical Reference(s):	SSRP, Chapter 1, Section 3.4.6, Area Designations		
Proposed references to be provided to applicants during examination:	None		
K/A Topic:	2.3.13 Knowledge of the radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.		
Question Source:	Bank. Diablo Canyon 2009 NRC Exam.		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.12/43.3/45.9/ 45.10		
Learning Objective:	L1525I09		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 73	Group #		4		
	K/A #		2.4.9 Knowledge of low power/shutdown implications in accident (e.g. loss of coolant accident or loss of residual heat removal) mitigation strategies.		
	Importance Rating		3.8		

Proposed Question:

Given the following plant conditions:

- Reactor Coolant System temperature is 120°F.
- Train 'A' RHR is in the shutdown cooling mode.
- Reactor vessel level is being lowered to facilitate Steam Generator nozzle dam installation.
- Reactor vessel level is minus 31 inches and decreasing.
- Pressurizer Relief Tank temperature, level and pressure are all increasing.
- The 'A' RHR pump is showing signs of cavitation.

What procedure should be implemented?

A. OS1201.02, RCS Leak

B. OS1201.10, Shutdown LOCA

C. OS1213.01, Loss of RHR During Shutdown Cooling.

D. OS1213.02, Loss of RHR While Operating at Reduced Inventory/Mid-Loop Conditions.

Proposed Answer: C

C is correct. The conditions in the stem of the question meet the entry conditions for OS1213.01, Loss of RHR During Shutdown Cooling, particularly signs of pump cavitation and increasing PRT level and pressure (which could be indicative of the RHR suction relief valve lifting). This procedure is applicable when RHR is in the shutdown cooling mode and reactor vessel level is above minus 36 inches.

A is incorrect but plausible. An increase in PRT temperature, level and pressure would occur if there were a loss of inventory from the RCS via the RHR suction relief valves. Additionally, if an RCS leak caused sufficient loss of inventory then RHR pump cavitation could occur. Procedure OS1201.02, RCS Leak does not contain mitigating strategies for the degradation of RHR pump or system performance while the RHR system is in the shutdown cooling mode.

B is incorrect but plausible. An increase in PRT temperature, level and pressure would occur if there were a loss of inventory from the RCS. Procedure OS1201.10, Shutdown LOCA is only applicable in Modes 3 (w/ SI Accumulators isolated) or Mode 4.

D is incorrect but plausible. The conditions in the stem of the question do support the symptoms or entry conditions for procedure OS1213.02, Loss of RHR While Operating at Reduced Inventory/Mid-Loop Conditions, particularly signs of pump cavitation and increasing PRT level and pressure (which could be indicative of the RHR suction relief valve lifting). This procedure is only applicable if Reactor Vessel level is below minus 36 inches. If the procedure were entered then step 1 of the procedure would redirect the operator to OS1213.01, Loss of RHR During Shutdown Cooling.

Technical Reference(s):	OS1213.01, Loss of RHR During Shutdown Cooling, Part A, Purpose and Part B, Symptoms or Entry Conditions.		OS1213.02, Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions, Part A, Purpose and Part B, Symptoms or Entry Conditions.
	OS1201.10, Shutdown LOCA, Part A, Purpose and Part B, Symptoms or Entry Conditions.		OS1201.02, RCS Leak, Part A, Purpose and Part B, Symptoms or Entry Conditions.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	2.4.9 Knowledge of low power/shutdown implications in accident (e.g. loss of coolant accident or loss of residual heat removal) mitigation strategies.

Question Source:	Bank. Teb 22017			
Question Cognitive Level:	Higher: Comprehension/Analysis			
10 CFR Part 55 Content:	41.10/43.5/45.13			
Learning Objective:	L1705I01, L1705I04, L1704I01, L1180I04			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 74	Group #		4		
	K/A #		2.4.11 Knowledge of abnormal condition procedures.		
	Importance Rating		4.0		

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- VAS alarm B5957, CONDENSATE PUMP DISCHARGE CONDUCTIVITY HIGH is in alarm.
- Chemistry has sampled the secondary system in accordance with procedure CD0905.07, Seawater In-Leakage.
- Chemistry has confirmed that there is a valid salt water intrusion and that Condensate Pump discharge conductivity is 1.5 micromhos.
- The crew has entered procedure OS1234.02, "Condenser Tube or Tube Sheet Leak."

What action should be taken?

A. Commence a power decrease to isolate the affected waterbox.

B. Trip the Reactor and go to procedure E-0, "Reactor Trip or Safety Injection".

C. Remain at 100% power and continue plant operation while monitoring the leak rate trend.

D. Commence a plant shutdown to Hot Standby per procedure OS1231.04, "Rapid Down Power".

Proposed Answer: B

B is correct. Procedure OS1234.02 contains continuous action step #7 which evaluates the need to trip the reactor. The threshold value for tripping the reactor is >1.0 micromho and that Chemistry has determined that there has been a valid salt water intrusion.

A is incorrect but plausible. If the Condensate Pump discharge conductivity is less than 1.0 micromho then the procedure directs performing a plant downpower per management recommendation in order to isolate the affected waterbox.

C is incorrect but plausible. If the Condensate Pump discharge conductivity is less than 1.0 micromho then the procedure contains the option of continuing plant operation per management recommendation and continuing to monitor leak rate trends.

D is incorrect but plausible. If the Condensate Pump discharge conductivity is less than 1.0 micromho then the procedure contains additional guidance for shutting the plant down to Hot

Standby per management recommendation.			
Technical Reference(s):	OS1234.02, Condenser Tube or Tube Sheet Leak		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	2.4.11 Knowledge of abnormal condition procedures.		
Question Source:	Bank. Seabrook 2003 Company Exam		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.10/43.5/45.13		
Learning Objective:	L1188I02, L1188I03		

Seabrook Station 2010 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #		3		
Question 75	Group #		4		
	K/A #		2.4.20 Knowledge of the operational implications of EOP warnings, cautions and notes.		
	Importance Rating		3.8		

Proposed Question:

Procedure E-0, "Reactor Trip or Safety Injection" contains an Operator Action Summary Page NOTE that directs tripping all Reactor Coolant Pumps if RCS Subcooling is less than 40°F. What is the basis for this action?

A. To prevent physical damage to the RCP casings associated with pumping a two-phase mixture.

B. To prevent secondary heat sink depletion by minimizing heat input into the Reactor Coolant System.

C. To minimize RCS inventory loss through a small break which may lead to core uncovery if the RCP's were tripped later in the accident.

D. To prevent damage to the RCP seal package due to the potential for a two-phase flow mixture existing within the pump casing.

Proposed Answer: C

C is correct. Per Westinghouse Owners Group ERG's, Generic Issue, RCP Trip/Restart "The reason for purposely tripping the RCP's during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCP's were tripped for some other reason later in the accident.

A is incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCP's is prudent. The document discusses tripping one RCP in procedure FR-C.2 to prevent pump damage due to running the pumps under two-phase/voided conditions. This situation is not applicable to the guidance specific to the 40°F subcooling criteria in E-0.

B is incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCP's is prudent. The document discusses tripping one RCP in procedure FR-H.1 to minimize secondary side inventory depletion. This situation is not applicable to the guidance specific to the 40°F subcooling criteria in E-0.

D is incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCP's is prudent. The document discusses tripping one RCP in procedure FR-C.2 to prevent pump damage due to running under the pumps under two-phase/voided conditions. This situation is not applicable to the guidance specific to the 40°F subcooling criteria in E-0.

Technical Reference(s):	E-0, Reactor Trip or Safety		Westinghouse Owners Group

	Injection, Operator Action Summary Page.		ERG's, Generic Issue, RCP Trip/Restart

Proposed references to be provided to applicants during examination:	None
K/A Topic:	2.4.20 Knowledge of the operational implications of EOP warnings, cautions and notes.
Question Source:	New
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.10/43.5/45.13
Learning Objective:	L1202I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 76	Group #				1
	K/A #		000007 Reactor Trip - Stabilization – Recovery 2.4.18 Knowledge of the specific bases for EOP's		
	Importance Rating		4.0		

Proposed Question:

Given the following plant conditions:

- The reactor has tripped.
- Safety Injection has actuated.
- The 'A' Main Steamline Radiation Monitor is in high alarm.
- The crew has entered E-0, "Reactor Trip or Safety Injection".
- The immediate action steps have been completed.
- Steam Generator 'A' level is 16% narrow range.
- Steam Generator 'B' level is 2% narrow range.
- Steam Generator 'C' level is 5% narrow range.
- Steam Generator 'D' level is 4% narrow range.

What action should the Unit Supervisor direct?

A. Maintain total EFW flow greater than 500 gallons per minute to maintain adequate heat sink.

B. Immediately isolate EFW flow to the 'A' Steam generator to prevent a generator overfill condition.

C. Isolate the Turbine Driven EFW pump steam supply from the 'A' Steam Generator to stop the unmonitored radioactive release.

D. Throttle EFW flow to the 'B', 'C' and 'D' steam generators. Isolate EFW flow to the 'A' Steam Generator when it's level is greater than 33% to establish thermal partitioning.

Proposed Answer: B

B is correct. The specific basis for isolating EFW flow to a ruptured steam generator is to prevent steam generator overfill conditions. Per SM 7.20, "Time Critical Actions, Figure 5.1, Time Critical Action (Validated), item 2, Terminate ECCS Break Flow to Prevent SG Overfill During a SGTR Event" EFW isolation is assumed to be performed per procedure E-0, Operator Action Summary Page such that the assumed level in the steam generator is no higher than 33%.

A is incorrect but plausible. Procedure E-0, step 19 directs maintaining EFW flow greater than 500 gpm until steam generator level criteria is met. In this case level criteria is met. Wide range levels

in the intact steam generators can be assumed to be greater than 65% as the narrow range levels are all on scale. C is incorrect but plausible. Isolation of the turbine driven EFW pump steam supply is a directed strategy for a ruptured steam generator and the specific basis is to isolate an unmonitored radioactive release, however this strategy is not directed in procedure E-0. The ruptured steam generator steam supply is isolated in procedure E-3, "Steam Generator Tube Rupture". D is incorrect but plausible. It is true that EFW flow can be throttled based on level criteria. It is also true that the ruptured steam generator level is verified based on establishing water level above the generator u-tubes to ensure partitioning. The level criteria for isolating EFW flow to a ruptured generator is 6% not 33%. The 33% value is a maximum assumed value in the basis for time critical action " Terminate ECCS break flow to prevent SG overfill during SGTR Event"

Technical Reference(s):	E-0, "Reactor Trip or Safety Injection"		SM 7.20, Time Critical Actions, Figure 5.1, Time Critical Action (Validated), item 2, Terminate ECCS Break Flow to Prevent SG Overfill During a SGTR Event

Proposed references to be provided to applicants during examination:	None

K/A Topic:	000007 Reactor Trip - Stabilization – Recovery
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.1/45.13
Learning Objective:	L1202I03, L1230I07

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 77	Group #				1
	K/A #		000015/17 RCP Malfunctions 2.1.20 Ability to interpret and execute procedure steps.		
	Importance Rating		4.6		

Proposed Question:

Given the following plant conditions:

- The plant is at 45% power.
- VAS alarm D5782, "RCP D Motor Frame Vibration High" is in alarm.
- Reactor Coolant Pump 'D' frame vibration is 4 mils and increasing at 1.0 mil per hour.
- Reactor Coolant Pump 'D' shaft vibration is 10 mils and increasing at 1.0 mil per hour.
- Vibration values have been determined to be valid.

What action is required?

A. Shutdown the plant to Mode 3 and then stop the 'D' RCP per OS1001.05, "Reactor Coolant Pump Operations."

B. Continue to monitor 'D' RCP vibration and notify Tech. Support per D5782, "RCP D Motor Frame Vibration High" alarm response procedure.

C. Feed the 'D' Steam Generator to between 60 and 70% NR level and trip the 'D' RCP in accordance with OS1201.01, "RCP Malfunction."

D. Trip the reactor and enter E-0 "Reactor Trip or Safety Injection." Stop the 'D' RCP after the immediate action steps are complete.

Proposed Answer: C

C is correct. The 'D' RCP frame vibration is above the alert value of "3 mils and increasing at greater then 0.2 mils per hour". Plant power is below the P-8 permissive value (50% power-reset @ 48%). In this case OS1201.01 directs feeding up the steam generators and removing the pump from service.

All of the distractors are plausible as they require application of procedure requirement knowledge and the understanding of the P-8 permissive relay.

A is incorrect. It is correct that the procedure directs a plant shutdown to Mode 3 however the RCP is removed from service first.

B is incorrect. If the frame vibration level was below the alert value then the abnormal procedure would direct continued monitoring of RCP parameters and notification of Tech. Support.

D is incorrect. If plant power were above P-8 (52%) then the procedure would direct tripping the reactor.

Technical Reference(s):	OS1201.01, "RCP Malfunction"		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	2.1.20 Ability to interpret and execute procedure steps.
Question Source:	Modified.2007 Seabrook Company Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.5/45.12
Learning Objective:	L1181I03

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 78	Group #				1
	K/A #		000025 Loss of RHR System AA2 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: AA2.04 Location and isolability of leaks.		
	Importance Rating				3.6

Proposed Question:

Given the following plant conditions:

- The crew has entered ECA-1.2, "LOCA Outside Containment" based upon RDMS indication of high radiation levels in the RHR vaults.
- RH-V-14, RHR Train A Discharge to RCS and RH-V-22, RHR Train A Cross-Connect have both been closed.
- Train 'A' RHR and CBS pumps have been placed in Pull-To-Lock.
- ECCS flow is stable.
- RCS pressure is 1100 psig and stable.

What action should be taken?

A. The LOCA is isolated. The crew should transition to E-1, "Loss of Reactor or Secondary Coolant."

B. The LOCA is isolated. The crew should transition to ES-1.1, "SI Termination."

C. The LOCA is NOT isolated. The crew should continue with ECA-1.2 procedure actions to try to identify and isolate the LOCA in Train 'B' RHR.

D. The LOCA is NOT isolated. The crew should transition to ECA-1.1, "Loss of Emergency Coolant Recirculation" to address potential loss of Train 'A' ECCS equipment.

Proposed Answer: C

C is correct. Step 2 of ECA-1.2 provides direction to isolate Train 'A' RHR from the RCS and then monitor RCS pressure to determine if the LOCA has stopped. If the LOCA is continuing then the procedure provides direction to realign Train 'A' RHR to the RCS and then perform the same actions for Train 'B' RHR. Step 2 of ECA-1.2 defines pressure "INCREASING DUE TO LEAK ISOLATION" as criteria for determining leak isolation. The conditions in the question stem indicate that RCS pressure is stable but not increasing which indicates that there is still an active leak.

A is incorrect but plausible. If the LOCA were isolated then the procedure would direct a transition to E-1, “Loss of Reactor or Secondary Coolant”.

B is incorrect but plausible. It is plausible that once the LOCA is isolated then SI termination would be performed by transitioning directly to ES-1.1, “SI Termination” however the procedural flowpath is to enter E-1. Additionally, the conditions in the stem of the question indicate that the LOCA is not isolated.

D is incorrect but plausible. It is true that the leak has not been isolated. It is plausible that there would be a procedural strategy to address Train “A” ECCS concerns by transitioning to ECA-1.1, however ECA-1.2 directs isolation of Train ‘B’ RHR to attempt to isolate the LOCA. If it is determined that the LOCA is still occurring after Train ‘B’ RHR is isolated then the procedure directs a transition to ECA-1.1.

Technical Reference(s):	ECA-1.2, “LOCA Outside Containment”.	
Proposed references to be provided to applicants during examination:	None	
K/A Topic:	000025 Loss of RHR System	
Question Source:	Modified. Teb 29959	
Question Cognitive Level:	Higher: Comprehension/Analysis	
10 CFR Part 55 Content:	43.5/45.13	
Learning Objective:	L1209I05	

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 79	Group #				1
	K/A #		038 Steam Generator Tube Rupture 2.4.6 Knowledge of EOP Mitigation Strategies		
	Importance Rating				4.7

Proposed Question:

Given the following plant conditions:

- A tube rupture has occurred in Steam Generator 'C'.
- Subsequently a loss of offsite power occurred.
- The crew has entered E-3, "Steam Generator Tube Rupture."
- ECCS flow has been terminated.
- Offsite power has been restored.
- The crew is performing step 38, "Evaluate RCP Status."

Which of the following describes the preferred course of action for operation of the Reactor Coolant Pumps, and why?

A. All available RCP's should be started to ensure uniform boron concentration.

B. No RCP's should be started. Starting any RCP will increase the rate of steam generator tube leakage.

C. NO RCP's should be started. Starting any RCP may cause ruptured steam generator safety valve actuation.

D. ONLY RCP 'C' should be started to provide pressurizer spray and minimize Pressurized Thermal Shock during cooldown.

Proposed Answer: D

D is correct. Per Westinghouse Background Document, E-3, "RCP operation is preferred during recovery from a steam generator tube rupture to provide normal pressurizer spray and to ensure homogeneous fluid temperatures and boron concentrations. In addition to minimizing pressurized thermal shock and boron dilution concerns this also aids in cooling the ruptured steam generator". The procedure step states that RCP 1C is the preferred pump as it is "best for sprays". If RCP 'C' cannot be started then the procedure directs starting all available RCP's to provide normal spray.

A is incorrect but plausible. It is true that one of the reasons for RCP restart is to ensure uniform boron concentration however the preferred method is to start RCP 'C' only.

B is incorrect but plausible. It is true that starting an RCP while on natural circulation will increase

the transfer of thermal energy into the steam generators. It is plausible that this could result in leakage through the ruptured steam generator tubes. The procedure includes a note describing this concern but does not prohibit restarting an RCP. C is incorrect but plausible. It is true that starting an RCP may cause a steam generator safety valve actuation. This would most likely occur with the specific steam generator associated with the RCP restarted. The procedure includes a note describing this concern but does not prohibit restarting an RCP.			
Technical Reference(s):	E-3, "Steam Generator Tube Rupture".		Westinghouse Background Document, E-3, Step 37, pgs 162-166
Proposed references to be provided to applicants during examination:			None
K/A Topic:	038 Steam Generator Tube Rupture		
Question Source:	Bank.		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.10/43.5/45.13		
Learning Objective:	L1205I02, L1205I03		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 80	Group #				1
	K/A #		055 Station Blackout EA2 Ability to determine or interpret the following as they apply to a Station Blackout: EA2.02 RCS cooling through natural circulation cooling to S/G cooling.		
	Importance Rating				4.6

Proposed Question:

Given the following plant conditions:

- A loss of all AC electrical power has occurred.
- The crew has entered ECA-0.0, "Loss of All AC Power" and is in the process of depressurizing the steam generators.
- During the steam generator depressurization the following conditions are noted:
  - ➢ Steam Generator Narrow Range Levels are 0%, 4%, 2% and 10% in 'A' through 'D' SG's respectively.
  - ➢ RCS Cold Leg cooldown rate is 75°F/hr.
  - ➢ Pressurizer level is offscale low.

What action is required and why?

A. Stop the cooldown and restore steam generator level to ensure adequate heat transfer capability.

B. Stop the cooldown and restore pressurizer level to prevent voiding in the reactor vessel upper head region.

C. Reduce the RCS cooldown rate to less than 50°F/hr to ensure that the RCP seals are cooled in a controlled manner.

D. Continue the cooldown and stop when steam generator pressures are LESS THAN 250 PSIG to prevent injection of accumulator nitrogen into the RCS.

Proposed Answer: D

D is correct. Per ECA-0.0, step 17, depressurization of the steam generators is stopped when steam generator pressures are LESS THAN 250 PSIG. Per the Westinghouse Background Document this value is based on the minimum SG pressure which prevents injection of accumulator nitrogen into the RCS, plus a margin of controllability to ensure that the depressurization limit is not violated. The depressurization limit is 125 psig.

All other conditions listed in the stem of the question support continuing the depressurization to 250 psig. The cooldown rate limit is 100°F/hr. The SG level criteria is 65% wide range in at least two SG's or 6% narrow range in at least one SG. Additionally, the procedure includes a note stating "pressurizer level may be lost and that upper head voiding may occur. Depressurization should not be stopped to prevent these occurrences.

A is incorrect but plausible. The SG depressurization does include level criteria however it requires 6% narrow range in at least one SG or 65% wide range in at least two. Level criteria is met.

B is incorrect but plausible. Depressurization of the steam generators may cause pressurizer level to go offscale low and may result in voiding in the vessel upper head. The procedure contains a note explaining this condition however the depressurization should not be stopped to prevent the condition from occurring.

C is incorrect but plausible. It is true that the cooldown rate must be limited in order to cool the RCP seals in a controlled manner however the cooldown rate is limited to 100°F/hr.

Technical Reference(s):	ECA-0.0, "Loss of All AC Power"		Westinghouse Background Document, ECA-0.0, pgs 115-126

Proposed references to be provided to applicants during examination:	None
K/A Topic:	055 Station Blackout
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	43.5/45.13
Learning Objective:	L8067I3, L8067I4, L8067I10

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 81	Group #				1
	K/A #		W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink EA2 Ability to operate and/or monitor the following as they apply to the Loss of Secondary Heat Sink: EA2.2 Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.		
	Importance Rating				4.3

Proposed Question:

Given the following plant conditions:

- The plant tripped from 100% power.
- The crew has transitioned to FR-H.1, "Response to Loss of Secondary Heat Sink."
- The motor driven AND steam driven EFW pumps tripped and cannot be restored.
- CCP 'A' is running.
- Pressurizer pressure is 2200 psig and increasing slowly.
- Steam Generator wide range levels are:
  - SG 'A': 22%
  - SG 'B': 24%
  - SG 'C': 12%
  - SG 'D': 32%
- Containment pressure is 2 psig and stable.

What action is required next?

A. Immediately initiate bleed and feed.

B. Depressurize SG's and feed with condensate pumps.

C. Try to establish start-up feedwater pump flow to SG's.

D. Do not establish feed flow to any SG. Consult with TSC.

Proposed Answer: A

A is correct. Per FR-H.1 if wide range level in any 3 SG's is less than 30% (51% for adverse containment) OR pressurizer pressure is greater than or equal to 2385 PSIG due to a loss of secondary heat sink, Steps 10 through 14 should be immediately initiated for bleed and feed. Westinghouse Background Document, FR-H.1, section 2.2, RCS Bleed and Feed Heat Removal states that operator action to establish bleed and feed heat removal can prevent or minimize core uncovery. The document discusses the effectiveness of bleed and feed being dependant on the timeliness of operator action to initiate bleed and feed following indications of the symptoms of loss of all secondary heat sink. It is essential to adhere to this procedural guidance to prevent the adverse effects of core uncovery.

B is incorrect but plausible. The procedure provides extensive guidance for establishing a feed source to the steam generators. The condensate pumps are one of the available feed sources however an attempt would be made to utilize the startup feedwater pump first. Additionally, the criteria for establishing bleed and feed are met.

C is incorrect but plausible. The procedure provides extensive guidance for establishing a feed source to the steam generators. The startup feedwater pumps are one of the available feed sources and would be the next procedurally driven source of feed however the criteria for establishing bleed and feed are met.

D is incorrect but plausible. The procedure does contain guidance to prevent establishing feed flow to a "hot dry steam generator" however a hot dry steam generator is defined as having less than 14% wide range level for non-adverse containment conditions. If containment conditions were adverse then the criteria is "less than 30%" wide range. Given the plant conditions the next action to take would be to immediately establish bleed and feed.

Technical Reference(s):	FR-H.1, "Response to Loss of Secondary Heat Sink"		Westinghouse Background Document, FR-H.1, Section 2.2, RCS Bleed and Feed Heat Removal, pg. 10 and 68.

Proposed references to be provided to applicants during examination:	None

K/A Topic:	W/E05 Inadequate Heat Transfer - Loss of Secondary Heat Sink			
Question Source:	Modified from bank. Teb 26636			
Question Cognitive Level:	Higher: Comprehension/Analysis			
10 CFR Part 55 Content:	43.5/45.13			
Learning Objective:	L1211I03			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 82	Group #				2
	K/A #		037 Steam Generator Tube Leak 2.4.11 Knowledge of abnormal condition procedures.		
	Importance Rating				4.2

Proposed Question:

Given the following plant conditions:

- A tube leak has been identified in the 'A' Steam Generator.
- The crew has entered OS1227.02, "Steam Generator Tube Leak."
- The operator is attempting to maintain Pressurizer level.
- Letdown flow has been isolated.
- The 'A' Charging Pump is delivering charging flow at maximum rate.
- The 'B' Charging Pump is in standby.
- Pressurizer level is 20% and decreasing.

What action should be taken next?

A. Start an additional charging pump and flow through the normal charging header.

B. Open SI-V-138 and SI-V-139, 'ECCS High Head Injection Valves' and start ECCS pumps as necessary.

C. Trip the reactor then actuate Safety Injection and go to E-0, "Reactor Trip or Safety Injection."

D. Align charging pump suction to the RWST and then go to E-0, "Reactor Trip or Safety Injection", step 1.

Proposed Answer: A

A is correct. Per OS1227.02, step 2b and OAS page item 1 pressurizer level should be maintained as follows:

1) Reduce letdown flow as necessary.
2) Increase charging flow and start a second charging pump as necessary.
3) If pressurizer level can NOT be maintained greater than 7% using two charging pumps through the normal charging header THEN perform the following:
    1) Trip the reactor.
    2) When the reactor trip is verified THEN actuate SI.
    3) Go to E-0, "Reactor Trip or Safety Injection", step 1.

The question stem states that one charging pump is running so the correct action to take next is to start the second charging pump.

B is incorrect but plausible. It is true that there may be a need for ECCS injection flow, particularly from the high head injection (charging) pumps. ECCS injection is not utilized unless pressurizer level cannot be maintained with two charging pumps through the normal charging header. If ECCS injection is warranted then the procedure directs tripping the reactor and actuating Safety Injection instead of manually aligning ECCS equipment. Manual alignment of ECCS equipment is a procedure action in E-3, "Steam Generator Tube Rupture" but not in OS1227.02.

C is incorrect but plausible. Tripping the reactor and actuating Safety Injection is part of the procedural strategy however that action is taken if pressurizer level cannot be maintained after starting a second charging pump.

D is incorrect but plausible. The procedure does include an action to align the charging pump suction to the RWST however this is done if the charging pump suction source (Volume Control Tank) level cannot be maintained, not the pressurizer level.

Technical Reference(s):	OS1227.02, Steam Generator Tube Leak.		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	037 Steam Generator Tube Leak
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.5/45.13
Learning Objective:	L1190I03, L1190I04

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 83	Group #				2
	K/A #		024 Emergency Boration AA2 Ability to determine and interpret the following as they apply to the Emergency Boration: AA2.02 When use of manual boration valve is needed.		
	Importance Rating				4.4

Proposed Question:

Given the following plant conditions:

- The crew is responding to a load rejection from 100% power.
- All systems respond as expected.
- D7762, "CONTROL BANK D INSERTION LIMIT LO-LO" is in ALARM.
- CS-V-426, "Emergency Borate to Charging Pump" will not open.

What action should the Unit Supervisor direct?

A. Align the Boric Acid Tank gravity feed to the Charging Pump suction per OS1202.04, "Rapid Boration."

B. Align the Refueling Water Storage Tank to the Charging Pump suction per OS1202.04, "Rapid Boration."

C. Start a boration using CVCS makeup controls at maximum rate per OS1231.04, "Rapid Downpower."

D. Start a boric acid pump and manually align CVCS makeup valves to the VCT per OS1008.01, "Chemical and Volume Control System Makeup Operations."

Proposed Answer: B

B is correct. Per OS1202.04, "Rapid Boration", step 2, the Response Not Obtained Action is to rapid borate using the Refueling Water Storage Tank. Step 3 of the procedure describes aligning the RWST to the charging pump suction.

A is incorrect but plausible. OS1202.04 includes detailed guidance for utilizing the Boric Acid Tank gravity lineup to the charging pumps however this action is utilized in Modes 4-6.

C is incorrect but plausible. It is plausible to perform a boration at maximum rate. This strategy would be utilized during a load rejection however it is not a strategy in OS1202.04.

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 83	Group #				2
	K/A #		024 Emergency Boration AA2 Ability to determine and interpret the following as they apply to the Emergency Boration: AA2.02 When use of manual boration valve is needed.		
	Importance Rating				4.4

Proposed Question:

Given the following plant conditions:

- The crew is responding to a load rejection from 100% power.
- All systems respond as expected.
- D7762, "CONTROL BANK D INSERTION LIMIT LO-LO" is in ALARM.
- CS-V-426, "Emergency Borate to Charging Pump" will not open.

What action should the Unit Supervisor direct?

A. Align the Boric Acid Tank gravity feed to the Charging Pump suction per OS1202.04, "Rapid Boration."

B. Align the Refueling Water Storage Tank to the Charging Pump suction per OS1202.04, "Rapid Boration."

C. Start a boration using CVCS makeup controls at maximum rate per OS1231.04, "Rapid Downpower."

D. Start a boric acid pump and manually align CVCS makeup valves to the VCT per OS1008.01, "Chemical and Volume Control System Makeup Operations."

Proposed Answer: B

B is correct. Per OS1202.04, "Rapid Boration", step 2, the Response Not Obtained Action is to rapid borate using the Refueling Water Storage Tank. Step 3 of the procedure describes aligning the RWST to the charging pump suction.

A is incorrect but plausible. OS1202.04 includes detailed guidance for utilizing the Boric Acid Tank gravity lineup to the charging pumps however this action is utilized in Modes 4-6.

C is incorrect but plausible. It is plausible to perform a boration at maximum rate. This strategy would be utilized during a load rejection however it is not a strategy in OS1202.04.

D is incorrect but plausible. It is plausible to perform a boration by manually starting a boric acid pump and aligning the associated CVCS boration flowpath. Additionally, step 4 of OS1202.04 includes a strategy for manually starting a boric acid pump. This overall strategy would not be utilized with the plant in Mode 1.			
Technical Reference(s):	OS1202.04, Rapid Boration		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	024 Emergency Boration		
Question Source:	New		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	43.5/45.13		
Learning Objective:	L1190I07		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 84	Group #				2
	K/A #		076 High Reactor Coolant Activity AA2 Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: AA2.02 Corrective actions required for high fission product activity in the RCS.		
	Importance Rating				3.4

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- RCS DOSE EQUIVALENT I-131 activity is 0.6 microCurie per gram and slowly rising.
- The crew has entered OS1202.05, "Reactor Coolant System High Activity."
- Chemistry has confirmed the high activity with a second RCS sample.
- Letdown filters and mixed bed demineralizers are aligned per Chemistry recommendations.
- Letdown flow has been maximized.

What action should be taken, if any?

A. No further action is required.

B. Place the plant in Hot Standby with Tavg less than 500°F within 6 hours.

C. Return to normal plant procedures AND perform isotopic analysis for Iodine once every 4 hours.

D. Evaluate effectiveness of letdown components and consult with Chemistry for improved cleanup methods.

Proposed Answer: D

D is correct. Step 4 of OS1202.05 provides guidance for evaluating Tech. Spec. actions. The RCS specific activity value for Dose Equivalent I-131 is below the Tech. Spec. 3.4.8 limit. There is no Tech. Spec. action required. Procedure step 5 then provides guidance for monitoring RCS activity. If the RCS activity is not decreasing then the Response Not Obtained states "Evaluate the effectiveness of letdown components and consult Chemistry and Tech. Support for improved cleanup methods.

A is incorrect but plausible. If the candidate correctly determines that there are no Tech. Spec.

actions required then they would eliminate distractors B and C. Additionally, the stem of the question includes some of the actions that have been performed per OS1202.05. The candidate could incorrectly determine that all of the actions that support RCS cleanup have been completed, in which case they may conclude that OS1202.05 has been completed.

B is incorrect but plausible. If RCS specific activity exceeded the Tech. Spec. 3.4.8 limit for microCurie per gram DOSE EQUIVALENT I-131 then distractor B would be correct. The value of DOSE EQUIVALENT I-131 is below the Tech. Spec. limit.

C is incorrect but plausible. It is plausible that 4 hour sampling would be required per Tech. Spec. 3.4.8 sampling requirements however the value of DOSE EQUIVALENT I-131 is below the Tech. Spec. limit.

	OS1202.05, "Reactor Coolant System High Activity"		Tech. Spec. 3.4.8, Reactor Coolant System Specific Activity.

Proposed references to be provided to applicants during examination:	None
K/A Topic:	076 High Reactor Coolant Activity
Question Source:	New.
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	43.5/45.13
Learning Objective:	L1181I09

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 85	Group #				2
	K/A #		W/E09&E10 Natural Circ. (E09) EA2 Ability to determine and interpret the following as they apply to the Natural Circulation Operations: EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency conditions.		
	Importance Rating				3.8

Proposed Question:

The operators are using ES-0.4, "Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS)", to perform a plant cooldown.

Plant conditions are as follows:

- RCS temperature is 450°F and stable.
- Pressurizer pressure is 900 psig and stable.
- Pressurizer level is 90% and stable.
- ECCS systems are disabled per procedure.
- Charging and letdown flows are matched.
- RCS subcooling is 84°F and stable.
- Subsequently, conditions to support RCP operation are established.

What action should be taken?

A. Start an RCP and transition back to ES-0.1, "Reactor Trip Response."

B. Do NOT start an RCP. Continue with ES-0.4, "Natural Circulation Cooldown."

C. Start an RCP and transition to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown."

D. Do NOT start an RCP and transition to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown."

Proposed Answer: C

C is correct. ES-0.4, step 1 is a continuous action step for establishing conditions for starting an RCP. Once an RCP is started the procedure directs a transition to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown"

A is incorrect but plausible. Procedure ES-0.4 is entered from ES-0.2, "Natural Circulation

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				1
Question 85	Group #				2
	K/A #		W/E09&E10 Natural Circ. (E09) EA2 Ability to determine and interpret the following as they apply to the Natural Circulation Operations: EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency conditions.		
	Importance Rating				3.8

Proposed Question:

The operators are using ES-0.4, "Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS)", to perform a plant cooldown.

Plant conditions are as follows:

- RCS temperature is 450°F and stable.
- Pressurizer pressure is 900 psig and stable.
- Pressurizer level is 90% and stable.
- ECCS systems are disabled per procedure.
- Charging and letdown flows are matched.
- RCS subcooling is 84°F and stable.
- Subsequently, conditions to support RCP operation are established.

What action should be taken?

A. Start an RCP and transition back to ES-0.1, "Reactor Trip Response."

B. Do NOT start an RCP. Continue with ES-0.4, "Natural Circulation Cooldown."

C. Start an RCP and transition to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown."

D. Do NOT start an RCP and transition to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown."

Proposed Answer: C

C is correct. ES-0.4, step 1 is a continuous action step for establishing conditions for starting an RCP. Once an RCP is started the procedure directs a transition to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown"

A is incorrect but plausible. Procedure ES-0.4 is entered from ES-0.2, "Natural Circulation

Cooldown". ES-0.2 may have been entered from ES-0.1, "Reactor Trip Response" if natural circ cooldown was desired. It may seem logical that a transition back to ES-0.1 would be directed however the procedure directs a transition out of the EOP network to OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown".

B is incorrect but plausible. It is plausible that if the cooldown/depressurization process has been initiated to the point where ECCS has been disabled that it would be preferable to continue with the cooldown process and not attempt to start an RCP. Restarting an RCP is one of the major actions of the procedure and is a "continuous action" step. An RCP should be started whenever one is available.

D is incorrect but plausible. It is plausible that the support conditions required for starting an RCP would include pressurizer level criteria to account for pump heat input however this is not the case. The procedure does have multiple steps that reference 90% pressurizer level criteria however this is for taking actions to control void growth and maintain pressurizer level control. The Westinghouse background document "RCP Trip/Restart" discusses pressurizer level with regard to RCP restart however it is to ensure adequate level is maintained in the pressurizer to account for void collapse upon pump start. The RCP restart step does check pressurizer level however the criteria is >65% vice <90%.

Technical Reference(s):	ES-0.4, "Natural Circulation Cooldown with Steam Void in Vessel (without RVLIS)"		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	W/E09&E10 Natural Circ.
Question Source:	Bank. Teb 25163
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	43.5/45.13
Learning Objective:	L1213I13

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 86	Group #				1
	K/A #		005 Residual Heat Removal A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: A2.03 RHR pump/motor malfunction.		
	Importance Rating				3.1

Proposed Question:

Given the following plant conditions:

- The crew has entered OS1201.10, “Shutdown LOCA.”
- The “A” RHR Pump is operating in the shutdown cooling mode.
- The “A” Charging Pump is operating; aligned to cold leg injection.
- RCS hot leg temperatures are 348°F.
- RCS pressure is 150 psig.
- Subcooling is approximately 20°F
- RWST level is 220,000 gallons and decreasing.

What action(s) should be taken?

A. Stop the RHR Pump(s) and place switches in Pull To Lock as per OS1201.10, “Shutdown LOCA.”

B. Enter ES-1.3, “Transfer to Cold Leg Recirculation.”

C. Actuate SI. Go to E-0, “Reactor Trip or Safety Injection.”

D. Start “B” Charging Pump and all available Safety Injection pumps as per OS1201.10, “Shutdown LOCA.”

Proposed Answer:	A	

KA match justification: This question tests the ability to determine that there may be a malfunction of the RHR pump due to inadequate suction head/cavitation. The question matches both parts of the two part KA statement as the question includes selecting the correct procedural guidance to correct, control or mitigate the consequences of the condition.

A is correct. Per OS1201.10 if Pressurizer level is less than 7% (28% for adverse containment) or Subcooling is less than 40°F the RHR pumps should be stopped and placed in Pull To Lock to prevent possible pump damage due to inadequate suction head/cavitation.

B is incorrect but plausible. OS1201.01 does have guidance for swapping over to Cold Leg Recirculation to maintain a suction source for the RHR/ECCS, however this is not done until RWST level is less than 115,000 gallons.

C is incorrect but plausible. If the plant were in Modes 1, 2 or 3 with accumulators aligned for injection and subcooling were less than 40°F then the correct action would be to actuate SI and go to E-0, "Reactor Trip or Safety Injection" however the conditions stated in the stem of the question indicate that the plant is in Mode 4. The guidance in OS1201.10 for inadequate subcooling is to stop the RHR pumps and perform subsequent steps to align ECCS pumps for injection into the RCS.

D is incorrect but plausible. There are steps later in the procedure for aligning ECCS pumps however the correct action given the conditions in the question stem is to stop the RHR pumps in order to protect the pumps.

Technical Reference(s):	OS1201.10, "Shutdown LOCA"
Proposed references to be provided to applicants during examination:	None
K/A Topic:	005 Residual Heat Removal
Question Source:	Bank. Teb 25131
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.12/ 45.13
Learning Objective:	L1704I02

Proposed Answer:	A	

KA match justification: This question tests the ability to determine that there may be a malfunction of the RHR pump due to inadequate suction head/cavitation. The question matches both parts of the two part KA statement as the question includes selecting the correct procedural guidance to correct, control or mitigate the consequences of the condition.

A is correct. Per OS1201.10 if Pressurizer level is less than 7% (28% for adverse containment) or Subcooling is less than 40°F the RHR pumps should be stopped and placed in Pull To Lock to prevent possible pump damage due to inadequate suction head/cavitation.

B is incorrect but plausible. OS1201.01 does have guidance for swapping over to Cold Leg Recirculation to maintain a suction source for the RHR/ECCS, however this is not done until RWST level is less than 115,000 gallons.

C is incorrect but plausible. If the plant were in Modes 1, 2 or 3 with accumulators aligned for injection and subcooling were less than 40°F then the correct action would be to actuate SI and go to E-0, "Reactor Trip or Safety Injection" however the conditions stated in the stem of the question indicate that the plant is in Mode 4. The guidance in OS1201.10 for inadequate subcooling is to stop the RHR pumps and perform subsequent steps to align ECCS pumps for injection into the RCS.

D is incorrect but plausible. There are steps later in the procedure for aligning ECCS pumps however the correct action given the conditions in the question stem is to stop the RHR pumps in order to protect the pumps.

Technical Reference(s):	OS1201.10, "Shutdown LOCA"		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	005 Residual Heat Removal
Question Source:	Bank. Teb 25131
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.12/ 45.13
Learning Objective:	L1704I02

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 87	Group #				1
	K/A #		006 Emergency Core Cooling 2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior and instrument interpretation.		
	Importance Rating				4.7

Proposed Question:

Given the following plant conditions:

- A small break LOCA has occurred.
- The crew is performing the actions of ES-1.2, "Post LOCA Cooldown and Depressurization."
- ECCS pumps have been stopped.
- Normal Charging is aligned.
- The crew is depressurizing the RCS.
- When the depressurization is stopped the following conditions exist:
  - RCS Subcooling is 37°F and decreasing.
  - Pressurizer Level is 18% and decreasing.

What action should be taken and why?

A. Manually start ECCS pumps as necessary to regain subcooling.
B. Increase RCS pressure using pressurizer heaters to regain subcooling.
C. Isolate letdown and check to ensure that Pressurizer level stabilizes above 7%.
D. Actuate Safety Injection and verify all safeguard equipment has actuated.

Proposed Answer: A

A is correct. ES-1.2, "Post LOCA Cooldown and Depressurization", step 22 and OAS page, Item 1, ECCS Reinitiation Criteria both direct manually aligning valves and starting ECCS pumps as necessary if RCS subcooling is less than 40°F or Pressurizer level is less than 7%.

B is incorrect but plausible. It is true that pressurizer heaters are operated during the depressurization process however this is done to establish saturated conditions in the pressurizer to maintain a steam bubble. The heaters are not used as part of the strategy to recover subcooling.

C is incorrect but plausible. The procedure does direct controlling charging flow as necessary to maintain Pressurizer level during the depressurization process. It is conceivable that charging and letdown could be utilized as part of the pressurizer level control strategy however letdown flow is not established in ES-1.2.

D is incorrect but plausible. It is true that the strategy to recover subcooling includes restarting ECCS equipment however a complete reinitiation of SI is not directed as this would result in a higher RCS pressure than necessary for the given plant conditions.

Technical Reference(s):	ES-1.2, "Post LOCA Cooldown and Depressurization"		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	006 Emergency Core Cooling
Question Source:	Bank. Seabrook 2000 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.12/45.13
Learning Objective:	L1204I06

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 88	Group #				1
	K/A #		012 Reactor Protection A2 Ability to (a) predict the impacts of the following malfunctions or operations on the RPS, and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: A2.05 Faulty or erratic operation of detectors and function generators.		
	Importance Rating				3.2

Proposed Question:

Given the following conditions:

- All actions for removing Power range channel N44 from service have been completed in accordance with OS1211.04, 'Power Range NI Instrument Failure.'
- Due to a seal problem, the crew has just performed a downpower to 47% and removed the 'D' Reactor Coolant Pump from service.
- 20 minutes later power range channel N41 begins to drift upward and is currently reading 52% power.

What action should be taken?

A. Verify reactor trip and enter E-0, 'Reactor Trip or Safety Injection.'

B. Declare power range channel N41 inoperable and within 1 hour make preparations to be in MODE 3 within the next 6 hours.

C. Bypass channel N41. Coordinate with I&C to troubleshoot channel N41. Place channel N41 associated bistables in the tripped condition within 6 hours.

D. Bypass channel N41. Do not trip channel N41 associated bistables. Immediately initiate corrective actions to return channel N41 or N44 to OPERABLE status.

Proposed Answer: A

A is correct. 2 power range channels are now above the P-8 setpoint (N41 and N44 with a tripped bistable). The reactor should trip on low RCS flow (1 of 4 loops). The reactor should have tripped.

B is incorrect but plausible. The Tech. Spec. actions for NI power range channels only cover the condition for 1 channel inoperable. If two were inoperable the crew should apply Tech. Spec. item 3.0.3. , however with power above P-8 the reactor should have tripped.

C is incorrect but plausible. This distracter describes the directed actions in OS1211.04 (step 2) if channel N41 were the only power range channel that was inoperable, however with power above P-8 the reactor should have tripped.

D is incorrect but plausible. With one channel inoperable step 2 of OS1211.04 directs not tripping additional bistables, as this would cause a reactor trip. There is no Tech. Spec. action that discusses taking immediate corrective actions to return the failed channel to service. This is plausible because Tech. Specs does have a similar statement for multiple failed components, as is the case with the Emergency Feedwater Tech. Spec. Additionally, this distracter is wrong because with power above P-8 the reactor should have tripped.

Technical Reference(s):	OS1211.04, "Power Range NI Instrument Failure".		Tech. Spec. item 3/4. 3.1, Reactor Trip System Instrumentation

Proposed references to be provided to applicants during examination:	None

K/A Topic:	012 Reactor Protection
Question Source:	Bank. Seabrook 2009 NRC Remediation Exam.
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.3/45.5
Learning Objective:	L1182I10

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 89	Group #				1
	K/A #		013 Engineered Safety Features Actuation A2 Ability to a)predict the impacts of the following malfunctions or operations on the ESFAS; and b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.06 Inadvertent ESFAS actuation.		
	Importance Rating				4.0

Proposed Question:

Given the following plant conditions:

- A Safety Injection has occurred.
- All equipment responded as designed.
- The crew has entered E-0, "Reactor Trip or Safety Injection."
- Steam Generator pressure boundaries are intact.
- Steam Generator u-tubes are intact.
- The RCS is intact.
- The E-0 procedural step for ECCS flow reduction is in progress.
- One CCP has been placed in standby and RCS pressure is stable.
- Pressurizer level is 96% and increasing.

What action should be taken?

A. Perform steps to terminate safety injection per ES-1.1, "SI Termination."

B. Reset the 'T' signal and establish letdown flow to reduce pressurizer level per ES-1.1, "SI Termination."

C. Align normal charging flow path and control charging flow to reduce pressurizer level per E-0", "Reactor Trip or Safety Injection."

D. Verify at least one thermal barrier cooling pump running, close CS-V-145, and stop ALL charging pumps per E-0, "Reactor Trip or Safety Injection."

Proposed Answer: D

D is correct. If Steam Generator pressure boundaries, Steam Generator u-tubes and the RCS are all intact then there was no reason to transition to procedures E-1, E-2 or E-3 and the Safety Injection does not appear warranted. E-0 has a continuous action step that directs stopping all CCP's if pressurizer level is greater than 95%. This step is in place to prevent pressurizer overfill and safety/relief challenges associated with an inadvertent safety injection. The step takes affect after the step for checking if ECCS flow can be reduced and stopping the first CCP. The conditions in the stem of the question indicate that the ECCS flow reduction step is in progress and that one CCP was secured w/RCS pressure stable. Given these conditions the continuous action step for securing all CCP's is applicable. The continuous action step directs securing the CCP's if pressurizer level is greater than 95%.

A is incorrect but plausible. It is plausible that the procedure may direct SI termination actions for a high pressurizer level. If pressurizer level were not greater than 95% then the procedural guidance does ultimately direct termination of SI utilizing ES-1.1. Termination of SI is the mitigation strategy associated with the time critical action for an inadvertent safety injection; however it is not the specific mitigation strategy for the E-0 continuous action step.

B is incorrect but plausible. It is plausible that establishing letdown flow would help mitigate a high pressurizer level condition. Resetting the 'T' signal and establishing letdown flow is part of the procedural strategy for recovering from the SI, however it would only be performed if pressurizer level were less than 95%. It is plausible that this action would be part of the ES-1.1 strategy for terminating SI and recovering inventory balance.

C is incorrect but plausible. It is plausible that aligning the normal charging flowpath and adjusting flow would mitigate the high pressurizer level condition. Establishing normal charging is part of the overall process of recovering from the SI; however it would only be performed if pressurizer level were less than 95%.

Technical Reference(s):	E-0, "Reactor Trip or Safety Injection, steps 12-16.		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	026 Containment Spray
Question Source:	New
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.3/45.13
Learning Objective:	L1202I07

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 90	Group #				1
	K/A #		061 Auxiliary/Emergency Feedwater 2.2.22 Knowledge of limiting conditions for operations and safety limits.		
	Importance Rating				4.7

Proposed Question:

Given the following plant conditions:

- The plant is in Mode 4.
- The Startup Feedwater Pump has been removed from service for a bearing replacement.
- The crew is preparing to enter MODE 3 to perform a reactor startup.

What ACTION must be taken (reference provided)?

A. Reactor startup CAN continue however the Startup Feedwater Pump must be OPERABLE before the plant enters MODE 1.

B. Reactor startup CAN continue however the plant CANNOT enter MODE 1 until performance of a risk assessment that determines the acceptability of entering MODE 1 has been completed.

C. Reactor Startup CANNOT commence. However the plant CAN enter MODE 3 after performance of a risk assessment that determines the acceptability of entering MODE 3.

D. The plant CANNOT enter MODE 3 until the Startup Feedwater Pump is OPERABLE.

Proposed Answer: D

This is an open reference question and is not "direct lookup". Per NUREG 1021, ES-602 an open reference question can be used to examine "how to apply information in a reference to the problem". This question tests the candidate's ability to utilize multiple sections of Tech. Specs. The candidate must utilize the specific information in the specification and apply it to memorized knowledge of Tech. Spec. item 3.0.4, applicability of when a Limiting Condition for Operation is not met.

D is correct. Tech. Spec. 3.7.1.2 specifically states that the provisions of Tech. Spec. item 3.0.4b do not apply for the startup feedwater pump. In this case the plant could not enter Mode 3 until the startup feedwater pump is OPERABLE and the specific limiting conditions for operation are met.

A is incorrect but plausible. If the specific NOTE in Tech. Spec. 3.7.1.2 that references LCO 3.0.4b is misapplied then the student may determine that the plant can ascend in Modes up to but not including Mode 1.

B is incorrect but plausible. If the specific NOTE in Tech. Spec. 3.7.1.2 is misapplied then the student may determine that the plant could enter Mode 1 after a risk assessment of "EFW" pumps is performed. The Startup Feedwater Pump is considered one of the three "auxiliary feedwater pumps" but is not considered an "EFW" pump as defined by the Tech. Spec. C is incorrect but plausible. If the specific NOTE in Tech. Spec. 3.7.1.2 that references LCO 3.0.4b is misapplied then the student may determine that the plant can enter Mode 3 after performance of a risk assessment.			
Technical Reference(s):	Tech. Spec 3.7.1.2, Auxiliary Feedwater System		Tech. Spec Bases, 3/4.7.1.2, Auxiliary Feedwater System and Bases.
	Tech. Spec.3/4.0, Applicability		
Proposed references to be provided to applicants during examination:			Tech. Spec Section 3.7, Auxiliary Feedwater System and bases. Provide whole section 3.7, Plant Systems
K/A Topic:	061 Auxiliary/Emergency Feedwater		
Question Source:	New		
Question Cognitive Level:	Higher: Comprehension/Analysis		
10 CFR Part 55 Content:	41.5/43.2/45.2		
Learning Objective:	L8010I13, L8045I09		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 91	Group #				2
	K/A #		002 Reactor Coolant 2.1.32 Ability to explain and apply system limits and precautions.		
	Importance Rating				4.0

Proposed Question:

Given the following plant conditions:

- Plant cooldown is in progress per OS1000.04, "Plant Cooldown from Hot Standby to Cold Shutdown."
- Both trains of RHR are aligned for shutdown cooling.
- Reactor Coolant Pump 'C' is running.
- Pressurizer level is stable at 75%.
- RCS Cold Leg temperatures are 230°F and decreasing.
- RCS pressure is 350 psig and stable.
- Pressurizer liquid and vapor temperatures are 430°F and stable.
- Pressurizer backup heaters are energized.
- One pressurizer spray valve is throttled open.
- Subsequently, the control board operator reports that pressurizer surge line temperature is 420°F and decreasing.

What plant condition exists and what procedural action should be taken?

A. A pressurizer insurge is in progress. The plant cooldown rate should be increased.
B. A pressurizer insurge is in progress. Letdown flow should be increased.
C. A pressurizer outsurge is in progress. Letdown flow should be decreased.
D. A pressurizer outsurge is in progress. The plant cooldown rate should be decreased.

Proposed Answer: B

B is correct. Procedure OS1000.04 contains extensive precautionary discussion regarding the need to limit a pressurizer insurge. Additionally, the procedure contains a contingency section to mitigate an insurge event. Decreasing pressurizer surge line temperature is indicative of a pressurizer insurge. The procedural contingency step directs the operators to INCREASE letdown flow and/or DECREASE charging flow in order to reestablish a constant pressurizer outflow.

A is incorrect but plausible. It is true that a pressurizer insurge is in progress however the procedurally driven strategy would be to increase letdown flow. Control of the plant cooldown rate is a valid procedural concern however it is associated with maintaining the overall plant cooldown rate to less than 100°F/hr and is not associated with the procedural strategy for mitigating a pressurizer insurge.

C is incorrect but plausible. It is true that adjusting letdown flow is part of the procedural strategy however it is no mitigate an insurge. A pressurizer outsurge is a desirable condition during the plant cooldown.

D is incorrect but plausible. It is true that a plant cooldown could contribute to a pressurizer outsurge as the reactor coolant specific volume decreases. A pressurizer outsurge is a desirable condition during the plant cooldown. Additionally, control of the plant cooldown rate is a valid procedural concern however it is associated with maintaining the overall plant cooldown rate to less than 100°F/hr and is not associated with the procedural strategy for mitigating a pressurizer insurge.

Technical Reference(s):	OS1000.04, "Plant Cooldown From Hot Standby to Cold Shutdown"		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	002 Reactor Coolant
Question Source:	Modified. Teb 26953
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.2/45.12
Learning Objective:	L1173I09

Seabrook Station 2010 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 92	Group #				2
	K/A #		033 Spent Fuel Pool Cooling A2 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: A2.02 Loss of SFPCS		
	Importance Rating				3.0

Proposed Question:

Given the following plant conditions:

- Core offload to the Spent Fuel Pool is complete.
- Computer alarms associated with a loss of Spent Fuel Pool (SFP) level have actuated.
- OS1215.07, "Loss of Spent Fuel Pool Cooling or Level" has been entered.
- The Unit Supervisor has determined that SFP level is 23.5 feet and decreasing rapidly.
- The leak location is currently unknown.

What actions should be taken?

A. Makeup from the Fire Protection System. Close the fuel transfer gate valve. Notify personnel in the Fuel Storage Building.

B. Makeup from the Demineralized Water System. Stop the SFP skimmer pump. Notify personnel in the Fuel Storage Building.

C. Makeup from CVCS. Notify HP control point. Stop the SFP skimmer pump. Monitor operation of the SFP cooling pumps.

D. Makeup from the Refueling Water Storage Tank. Notify HP control point. Stop the SFP skimmer pump. Shutdown the SFP cooling pumps.

Proposed Answer: D

D is correct. OS1215.07 has a specific strategy if SFP level is less than 25.4 feet and decreasing rapidly. This condition is such that SFP cooling system capability is compromised. The skimmer pump suction is located at the 25.6 ft. level and the cooling pump suction strainer is located at 23.4

ft. The procedural strategy includes establishing emergency makeup to the SFP from the RWST and securing both the pool skimmer pump and the cooling pumps. Additionally, notifications are made to the HP checkpoint and to personnel inside the Fuel Storage Building.

A is incorrect but plausible. It is true that the procedure prescribes emergency makeup however closure of the transfer gate valve is only directed if the SFP leak is later determined to be located in the cask handling or fuel transfer area.

B is incorrect but plausible. It is true that the skimmer pump is stopped and that personnel in the building are notified. Makeup to the SFP from Demin Water is only an option if pool level is not decreasing rapidly.

C is incorrect but plausible. It is true that the skimmer pump should be secured however the cooling pumps should be secured as well vice monitoring their operation. It is plausible that the procedure would direct monitoring the cooling pump operation as level stated in the question stem is above the pump suction strainer. Additionally, it is true that HP should be notified. Makeup to the SFP from CVCS is only an option if pool level is not decreasing rapidly.

Technical Reference(s):	OS1215.07, "Loss of Spent Fuel Pool Cooling or Level"		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	033 Spent Fuel Pool Cooling
Question Source:	Modified from bank. Seabrook 2007 Company Exam
Question Cognitive Level:	Memory or Fundamental Knowledge
10 CFR Part 55 Content:	41.5/43.5/45.3/45.13
Learning Objective:	L1191I07, L1192I08

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				2
Question 93	Group #				2
	K/A #		016 Non-Nuclear Instrumentation System (NNIS) A2 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: A2.01 Detector failure.		
	Importance Rating				3.1

Proposed Question:

Given the following plant conditions:

- The plant is at 75% power.
- All control systems are in automatic.
- The controlling Pressurizer Pressure channel slowly fails high.
- The PSO reports pressure is 1940 psig and slowly decreasing due to an open spray valve.
- The US enters OS1201.06, "PZR Pressure Instrument/Component Failure."
- Spray valve RC-PCV-455A cannot be closed by any means.

Per OS1201.06, what action is required?

A. Trip "C" RCP and then trip the reactor. Enter E-0, "Reactor Trip or Safety Injection."
B. Commence down-power per OS1231.04, "Rapid Downpower." Raise charging flow to compress the PZR bubble. Trip "C" RCP when power is less than P-8.
C. Trip the reactor. Complete immediate actions of E-0, "Reactor Trip or Safety Injection." Trip "C" RCP and up to two more RCP's, as necessary.
D. Trip the reactor. Actuate SI. Enter E-0, "Reactor Trip or Safety Injection." Concurrently trip "C" RCP and up to two more RCP's, as necessary.

Proposed Answer: C

C is correct. Per OS1201.06 if the 'A' spray valve cannot be closed then the reactor should be tripped. After the immediate actions of procedure E-0 are complete then OS1201.06 should be utilized in parallel with action taken to trip the 'C' RCP. If pressure is still decreasing then additional an additional two RCP's should be stopped as necessary.

A is incorrect but plausible. It is true that the procedural action is to trip the RCP in order to mitigate the pressure transient however the correct sequence is to trip the reactor first and then trip the reactor coolant pump once the immediate actions of E-0 have been completed.

B is incorrect but plausible. Raising charging flow would have the effect of compressing the pressurizer steam space and would counteract the decrease in pressure but only to a marginal degree. Raising charging flow is not part of the strategy. It is also plausible that plant power would be reduced expeditiously and that the 'C' RCP would be secured when power is below P-8.

D is incorrect but plausible. It is true that the actions would include tripping the reactor and tripping the 'C' RCP and additional RCP's if necessary however the pressure decrease would not warrant a Safety Injection. Actuating Safety Injection is not part of the strategy.

Technical Reference(s):	OS1201.06, "PZR Pressure Instrument/Component Failure"		

Proposed references to be provided to applicants during examination:	None
K/A Topic:	016 Non-Nuclear Instrumentation System (NNIS)
Question Source:	Bank.Teb 26984
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.5/43.5/45.3/45.5
Learning Objective:	L1182I05

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 94	Group #				
	K/A #		2.1.23 Ability to perform specific system and integrated plant procedures during all modes of operation.		
	Importance Rating				4.4

Proposed Question:

The following plant conditions exist:

- The reactor has tripped and safety injection has actuated.
- While performing E-1, "Loss of Reactor or Secondary Coolant," an ORANGE path condition was noted for the Core Cooling critical safety function.
- FR-C.2, "Response to Degraded Core Cooling" was entered in response to this condition.
- While performing the steps of this procedure, the crew notes that valid RED path conditions exist for **BOTH** the Heat Sink and Containment critical safety functions.

Based on these conditions, what course of action should the Unit Supervisor take?

A. Stop performing FR-C.2, and immediately address the Heat Sink RED path.

B. Stop performing FR-C.2, and immediately address the Containment RED path.

C. Complete the actions of FR-C.2, and then address the Heat Sink RED path.

D. Complete the actions of FR-C.2, and then address the Containment RED path.

Proposed Answer: A

A is correct. Per OP 9.2, Emergency Operators Users Guide, Section 4.3, Control Room Usage of Status Trees, the order of priority for critical safety functions is Subcriticality, Core Cooling, Heat Sink, Integrity, Containment, Inventory, Emergency Recirculation, and RDMS. The order of severity priority is Red, Orange, Yellow, and Green. If any Orange terminus is encountered, the operator is expected to monitor all of the remaining trees, if no Red terminus is present, then suspend any ERP or ECA and address the Orange condition. If during the performance of an Orange condition FRP, any Red condition or higher priority Orange arises, then the higher priority condition should be addressed and the original Orange FRP is suspended.

B is incorrect but plausible. A Containment Red path is high priority however the rules of status tree usage dictate that it is of less priority than Heat Sink.

C is incorrect but plausible. An Orange path procedure is a high priority however the rules of status tree usage dictate that if a Red priority occurs then the Orange FRP should be suspended.

D is incorrect but plausible. An Orange path procedure is a high priority however the rules of status tree usage dictate that if a Red priority occurs then the Orange FRP should be suspended.

Technical Reference(s):	OP 9.2, Emergency Operators Users Guide, Section 4.3, Control Room Usage of Status Trees		

Proposed references to be provided to applicants during examination:	None

K/A Topic:	2.1.23 Ability to perform specific system and integrated plant procedures during all modes of operation.

Question Source:	Bank. Seabrook 2007 NRC Exam			

Question Cognitive Level:	Higher: Comprehension/Analysis		

10 CFR Part 55 Content:	41.10/43.5/45.2/45.6	

Learning Objective:	L1195I05

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 95	Group #				
	K/A #		2.1.36 Knowledge of procedures and limitations involved in core alterations.		
	Importance Rating				4.1

Proposed Question:

OS1000.09, Refueling Operation, requires that no more than two irradiated fuel assemblies be allowed in the cavity and canal at any one time.

Which of the following statements describes how this limitation is applied?

A. Fuel in the Transfer Car is counted as if it were in the canal until it is latched by the Spent Fuel Handling Tool.

B. Fuel in the cavity is counted as in the core as soon as it is being lowered to its core location.

C. Fuel is counted as in the canal until it is unlatched at its storage location in the spent fuel pool.

D. Fuel in the core is counted as in the cavity as soon as it is above and clear of the vessel.

Proposed Answer: A

A is correct. Per OS1000.09, "Refueling Operation", Figure 1: Limitations and Setpoints, fuel in the Transfer Car is counted as if it were in the canal until it is latched by the Spent Fuel Handling Tool.

B and D are incorrect but plausible. Each of the distractors describes a situation when a fuel assembly is in a transitional location with respect to the core/cavity. Each distractor could be misinterpreted as a valid defining boundary for the cavity.

C is incorrect but plausible. The statement is conservative with respect to the canal similar to the wording of the procedural requirement "Fuel in the core is counted as in the cavity as soon as it is latched with the manipulator crane".

Technical Reference(s):	OS1000.09, "Refueling Operation"		

Proposed references to be provided to applicants during examination: None

K/A Topic:	2.1.36 Knowledge of procedures and limitations involved in core alterations.			
Question Source:	Bank. Seabrook 2000 NRC Exam			
Question Cognitive Level:	Memory or Fundamental Knowledge			
10 CFR Part 55 Content:	41.10/43.6/45.7			
Learning Objective:	No specific facility objective			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 96	Group #				
	K/A #		2.2.21 Knowledge of pre and post maintenance operability requirements.		
	Importance Rating				4.1

Proposed Question:

Given the following plant conditions:

- The plant is in Mode 1.
- Significant boric acid deposits exist on the valve body of Containment Isolation Valve SI-V-70.
- Station management has determined that the boric acid deposits should be removed in order to assess valve packing leak rates.
- Cleaning of the MOV will be performed per a work order.

What post maintenance retesting is required, if any, for SI-V-70 after the boric acid cleaning is complete? (Reference provided)

A. No retesting is required.
B. An IST stroke time retest and a Containment Isolation Valve stroke test are required.
C. A full stroke exercise test, Containment Isolation Valve stroke test and a Position Indication Verification test are required.
D. A full stroke exercise test, Containment Isolation Valve stroke test, Position Indication Verification test, and a LLRT (local leak rate test) are required.

Proposed Answer: A

This is an open reference question and is not "direct lookup". Per NUREG 1021, ES-602 an open reference question can be used to exam "how to apply information in a reference to the problem". This question tests the candidates ability to utilize multiple sections of procedure MA3.5 in order to determine the need for any retest requirements.

A is correct. Per MA3.5, Figure 5.4, Post Maintenance Testing Guide, Item 1, Motor Operated Valves, no retesting is required "following painting, cleaning, or grease inspection or other cosmetic maintenance of the MOV".

Distractors B, C and D are all incorrect but plausible as they are described in MA3.5, Figure 5.1, In-Service Test Program Valve List, as being tests associated with SI-V-70 if testing were required.

Technical Reference(s):	MA3.5, Post Maintenance Testing		

Proposed references to be provided to applicants during examination:					MA3.5, Post Maintenance Testing
K/A Topic:	2.2.21 Knowledge of pre and post maintenance operability requirements.				
Question Source:	Bank. Teb 30156				
Question Cognitive Level:	Memory or Fundamental Knowledge				
10 CFR Part 55 Content:	41.10/43.2				
Learning Objective:	L1514I03				

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 97	Group #				
	K/A #		2.2.40 Ability to apply Technical Specifications for a system.		
	Importance Rating				4.7

Proposed Question:

Given the following plant conditions:

- 100% power.
- No radioactive release is in progress.
- Wide Range Gas Monitor (WRGM) conditions:
  - CP-434 Heat Trace circuit 28 tripped 45 minutes ago.
  - CP-426 Heat Trace circuit 46 is now energized.
  - Outside ambient temperature is 45 °F.
  - Inside ambient temperature is 66 °F.
  - All other WRGM equipment is operable.
  - All associated alarms are reset.

What ACTION is required, if any? (Reference material provided)

A. Enter Action 36. Obtain dew point measurements every 12 hours.

B. Enter Action 36. No additional action is required because CP-426 circuit 46 is energized.

C. Declare WRGM inoperable when CP-434 circuit 28 tripped until loss of heat trace has been evaluated. Enter Action 32, 33, 35, and 36.

D. No action is required. WRGM sample line heat trace is not required if inside ambient temperature is greater than or equal to 20 °F above outside ambient temperature.

Proposed Answer: B

This is an open reference question and is not "direct lookup". Per NUREG 1021, ES-602 an open reference question can be used to exam "how to apply information in a reference to the problem". This question tests the candidates ability to utilize multiple sections of ODCM 5.2 in order to determine the required action. The question is not a direct lookup as there are three components (sample line ΔT, heat trace circuitry and dewpoint measurements) that must be analyzed in order to determine WRGM operability and/or the associated ACTION ITEM requirements. The student must make use of all the information in the ODCM specification to

determine a) if the WRGM is operable, b) if associated ACTION 36 should be entered and/or c) if dew point measurements must be taken per ACTION 36. The student must utilize the associated ODCM information associated with heat tracing to determine that Action 36 must be entered however the heat trace action has been met.

B is correct because ODCM Table A.5.2-1 item 2F Action 36 but page A.5-14 states that action 36 is satisified if CP-426 circuit 46 is energized within one hour.

A is incorrect but plausible: Only required if sample line temperature is < 20 degrees from outside ambient. Sample line temp alarms are reset .

C is incorrect but plausible: Page A.5-14 states that action 36 is satisfied and WRGM **remains** operable if circuit 46 is energized within one hour.

D is incorrect but plausible: Sample line temperature not inside ambient is required for determination of required actions for sample line temperature.

Technical Reference(s):	ODCM, Section 5.2, Radioactive Gaseous Effluent Monitoring Instrumentation		

Proposed references to be provided to applicants during examination:	ODCM (Provide whole document)

K/A Topic:	2.2.38, Knowledge of conditions and limitations in the facility license.

Question Source:	Bank. Teb 30629			
Question Cognitive Level:	Higher: Comprehension/Analysis			
10 CFR Part 55 Content:	41.10/43.2/43.5/45.3			
Learning Objective:	L1512I06			

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 98	Group #				
	K/A #		2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.		
	Importance Rating				3.7

Proposed Question:

The following plant conditions exist:

- Refueling outage is in progress.
- A task is going to be performed in the "A" Steam Generator.
- A Planned Special Exposure (PSE) is going to be used for an individual employee.
- The PSE has been justified as it will reduce the collective doses of all the personnel working on the task.

What reviews and approvals are required when processing the PSE?

A. Health Physics Department Supervisor, Health Physics Department Manager, Operations Manager.

B. Health Physics Department Supervisor, Health Physics Department Manager, Station Director.

C. Shift Manager, Health Physics Department Manager and the Station Director.

D. Health Physics Department Manager and the Station Director. If time permits, obtain verification that the NRC regional office has reviewed the Planned Special Exposure

Proposed Answer: D

D is correct per RP 5.2, Planned Special Exposures.

A is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed, however it is from the Station Director vice the Operations Manager.

B is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed from the Station Director . There is no approval required from the HP Supervisor. HP Supervisor approval is for lower administrative dose approval.

C is incorrect but plausible. The planned special exposure does require approval of the HP Dept. Manager. Additional station management approval is needed from the Station Director . There is no

requirement for approval from the Outage Containment Coordinator.			
Technical Reference(s):	RP5.2, Planned Special Exposures		
Proposed references to be provided to applicants during examination:			None
K/A Topic:	2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.		
Question Source:	Bank. Seabrook 2007 NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.12/43.4/45.10		
Learning Objective:	L1525I, Objective L1525I13		

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 99	Group #				
	K/A #		2.4.5 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.		
	Importance Rating				4.3

Proposed Question:

Given the following plant conditions:

- Reactor trip and Safety Injection from 100% power.
- Steam Generator pressures:
  - SG 'A': 150 psig and rapidly decreasing.
  - SG 'B': 800 psig and slowly decreasing.
  - SG 'C': 950 psig and slowly increasing.
  - SG 'D': 930 psig and slowly decreasing.
- Steam Generator levels:
  - SG 'A': 0% narrow range.
  - SG 'B': 4% narrow range and slowly increasing.
  - SG 'C': 25% narrow range and rapidly increasing.
  - SG 'D': 8% narrow range and slowly increasing.
- Main Steamline Radiation Monitors:
  - SG 'A': 0.31 mR/hr and stable.
  - SG 'B': 0.38 mR/hr and stable.
  - SG 'C': 1.2 E+2 mR/hr and increasing.
  - SG 'D': 0.34 mR/hr and stable.
- No Red or Orange path Critical Safety Function Status indications.

The proper procedure flowpath for this event will be E-0, "Reactor Trip or Safety Injection" to…

A. E-2, "Faulted Steam Generator Isolation" to E-3, "Steam Generator Tube Rupture."

B. E-2, "Faulted Steam Generator Isolation" to E-3, "Steam Generator Tube Rupture" to ECA-3.1, "SGTR With Loss Of Reactor Coolant-Subcooled Recovery Desired."

C. E-3, "Steam Generator Tube Rupture" to E-2, "Faulted Steam Generator Isolation."

D. E-3, "Steam Generator Tube Rupture" OAS page transition to E-2, "Faulted Steam Generator Isolation" and then back to E-3, "Steam Generator Tube Rupture."		
Proposed Answer:	A	

A is correct. The E-0 diagnostic step order is E-2, E-3, E-1. The flowpath would be from E-0 to E-2 first and then to E-3.

B is incorrect but plausible. There is a faulted SG however it is not the ruptured SG so a transition to ECA-3.1 is not correct.

C is incorrect but plausible. It is true that the rupture and fault conditions will be addressed however E-3 is performed after the faulted SG is isolated in E-2.

D is incorrect but plausible. There is a transition to E-2 from the E-3 Operator Action Summary Page however this would mean that an incorrect transition was made from E-0 to E-3.

Technical Reference(s):	E-0, "Reactor Trip or Safety Injection		E-2, "Faulted Steam Generator Isolation"
	E-3, "Steam Generator Tube Rupture"		ECA-3.1, "SGTR With Loss of Reactor Coolant-Subcooled Recovery Desired"

Proposed references to be provided to applicants during examination:	None
K/A Topic:	2.4.5 Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.
Question Source:	Bank. Diablo Canyon 2007 NRC Exam
Question Cognitive Level:	Higher: Comprehension/Analysis
10 CFR Part 55 Content:	41.10/43.5/45.13
Learning Objective:	L1202I09, L1205I02, L1205I03, L1207I02

Examination Outline Cross-reference:	Level		RO		SRO
	Tier #				3
Question 100	Group #				
	K/A #		2.4.27 Knowledge of fire in the plant procedures.		
	Importance Rating				3.9

Proposed Question:

Given the following plant conditions:

- A fire has occurred in the plant.
- The crew has entered OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room."
- The crew is in the process of establishing safe shutdown conditions and manually trips the reactor.

What action should be taken next?

A. Continue with actions in OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room". Procedure E-0, "Reactor Trip or Safety Injection" is NOT implemented.

B. Immediately transition to E-0, "Reactor Trip or Safety Injection", perform the immediate actions and then return to OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room."

C. Continue with actions in OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room." A transition will be made to Procedure E-0, "Reactor Trip or Safety Injection" once safe shutdown conditions are established.

D. Immediately transition to E-0, "Reactor Trip or Safety Injection." ES-0.1, "Reactor Trip Response" will direct a transition back to OS1200.01, "Safe Shutdown and Cooldown From the Main Control Room."

Proposed Answer: A

A is correct. A "Note" prior to step 1 of OS1200.01 states "E-0 should not be entered when the reactor is tripped in this procedure". Procedure step 1b provides guidance for verifying that the reactor is tripped.

B is incorrect but plausible. It is plausible that the E-0 immediate actions would be utilized in tandem with abnormal procedure OS1200.01. Procedural rules of usage allow for parallel AOP procedure use with EOPs.

C is incorrect but plausible. It is plausible that E-0 would be utilized in tandem with abnormal procedure OS1200.01. Procedural rules of usage allow for parallel AOP procedure use with EOPs.

D is incorrect but plausible. It is plausible that the procedural rules of usage would dictate entering the EOP network due to a valid entry condition (reactor trip) and that ES-0.1, "Reactor Trip

Response" would include a transition our of the EOP network to OS1200.01 however this is not the case.			
Technical Reference(s):	OS1200.02, "Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities".		
Proposed references to be provided to applicants during examination:	None		
K/A Topic:	2.4.27 Knowledge of fire in the plant procedures.		
Question Source:	New		
Question Cognitive Level:	Memory or Fundamental Knowledge		
10 CFR Part 55 Content:	41.10/43.5/45.13		
Learning Objective:	L8210I07		