Examination Outline Cross-reference:	Level	RO	SRO
Q1	Tier #	1	
	Group #	1	
	K/A #	**007** Reactor Trip **EK1** Knowledge of the operational implications of the following concepts as they apply to the reactor trip: **EK1.04** Decrease in reactor power following reactor trip (prompt drop and subsequent decay)	
	Importance Rating	3.6	

Proposed Question:

The Plant was operating at steady state full power when a loss of off-site power occurred. The following indications were observed during the performance of Step 1 of E-0, "Reactor Trip or Safety Injection":

- Neutron flux is 4%.
- Intermediate Range Startup Rate is -.5 decades per minute.
- All rod bottom lights are lit with the exception of Rod H8 which indicates full out.
- RTB is closed and cannot be opened.
- RTA is open.

Which one of the following actions should the crew take, and why?

A. Go to FR-S.1, "Response to Nuclear Power Generation/ATWS" because one reactor trip breaker remains closed.

B. Continue in E-0, "Reactor Trip or Safety Injection", indications validate that the reactor is tripped.

C. Go to FR-S.2, "Response to Loss of Core Shutdown" because startup rate does not meet reactor trip criteria.

D. Go to FR-S.1, "Response to Nuclear Power Generation/ATWS" because all control rods are not fully inserted.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. With one of the reactor trip breakers not open it is plausible the reactor is not tripped and actions in FR-S.1 are required to shut down the reactor.

B. Correct. In E-0 reactor trip is verified by Rod bottom lit, Reactor trip breakers open and Neutron flux decreasing. The indications given in the stem of the question are that the reactor is tripped.

C. Incorrect but plausible. Intermediate flux rate of more negative than -.2 DPM is required in the Subcriticality CSF tree for the tree to be SAT. If it is not then FR-S.2 is the procedure recommended by the CSF tree.

D. Incorrect but plausible. It is plausible that since not all rod bottom lights are lit the reactor is not tripped and actions of FR-S.1 are required to shut down the reactor.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection (Rev 50)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1202I03RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**10**	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q2	Tier #	1	
	Group #	1	
	K/A #	**008** Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) **AA1.** Ability to operate and / or monitor the following as they apply to the Pressurizer Vapor Space Accident: **AA1.02** HPI pump to control PZR level/pressure	
	Importance Rating	4.1	

Proposed Question:

An automatic Reactor Trip and Safety Injection have occurred as a result of lowering RCS pressure.

The operators note the following conditions:

- Pressurizer pressure dropping prior to and following the SI
- RCS average temperature stable prior to and following the SI
- Pressurizer level 30% and rising prior to the SI
- Pressurizer level 70% and rising 3 minutes following SI

Based on noted conditions how will charging pumps be operated and what caused the SI?

A. Stop ALL CCPs — Small Break LOCA from Cold Leg

B. Maintain CCP operating — Small Break LOCA from Cold Leg

C. Stop ALL CCPs — Small Break LOCA from PZR Vapor Space

D. Maintain CCP operating — Small Break LOCA from PZR Vapor Space

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is plausible that both CCPs should be secured. The student will see from the stem of the question that pressurizer level was 70% and rising at a fast rate after SI was initiated. Given the concern for pressurizer overfill, the student may decide that it is prudent to secure the CCPs, however this is not the case (see answer D statement). Additionally, it is plausible that the plant conditions could be a small break cold leg LOCA, as this would also result in a pressure drop prior to SI initiation, and a pressure drop for a finite period of time

after SI is initiated, however in the event of a small break cold leg LOCA, pressurizer level would not rise prior to SI initiation.

B. Incorrect but plausible. It is true that both CCPs should remain operating (see answer D statement). It is plausible that the plant condition could be a small break cold leg LOCA, as this would also result in a pressure drop prior to SI initiation, and a pressure drop for a finite period of time after SI is initiated, however in the event of a small break cold leg LOCA, pressurizer level would not rise prior to SI initiation.

C. Incorrect but plausible. It is true that the plant condition would be a small break steam space LOCA. It is plausible that both CCPs should be secured. The student will see from the stem of the question that pressurizer level was 70% and rising at a fast rate after SI was initiated. Given the concern for pressurizer overfill, the student may decide that it is prudent to secure the CCPs, however this is not the case (see answer D statement).

D. **Correct.** It is true that both CCPs should be operating. Given the conditions in the stem of the question , the procedural flowpath would lead to E-0, “Reactor Trip or Safety Injection”, step 11 to check if the RCS is intact. Given that this is a steam space LOCA, containment radiation or containment pressure may or may not be elevated, either by a release from the PRT rupture disc or a steam space pipe/PZR vessel break. If these conditions exist at step 11 there would be a procedural transition to E-1, “Loss of Reactor or Secondary Coolant”. If neither of those indications were present then the procedure flowpath would lead to E-0, step 21, to check PRT conditions. If the steam space break were from a PORV or PZR safety valves, then at step 21 there would also be a procedure transition to E-1. In either case the contingency step of E-0 for securing all CCPs due to high pressurizer level would have been bypassed, this level criteria for securing the CCPs is >95% level.

Additionally, the question stem states that pressurizer level was rising prior to SI, and that it continued to rise at a rapid rate after SI was initiated. These conditions are indicative of a vapor space LOCA.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection (Rev 50)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1203I02RO, 03RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure				

to provide the information will necessitate a detailed review of every question.)			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**10**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q3	Tier #	1	
	Group #	1	
	K/A #	**009** Small Break LOCA **EK3** Knowledge of the reasons for the following responses as the apply to the small break LOCA: **EK3.03** Reactor trip and safety initiation	
	Importance Rating	4.1	

Proposed Question:

Plant conditions:

- An RCS leak is occurring.
- The crew is performing actions of OS1201.02, “RCS Leak”.
- The leak is slowly increasing in size.

At what point will a manual reactor trip and safety injection initiation be required?

A. RCS pressure decreases to 1945 psig.
B. VCT level cannot be maintained >15%.
C. PZR level cannot be maintained >7% with one charging pump.
D. PZR level cannot be maintained >7% with two charging pumps.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. The “Pressurizer Low Pressure” reactor trip setpoint is 1945 psig. If the reactor trip setpoint were reached and an automatic reactor trip did not occur then a manual reactor trip would be warranted, however the SI actuation setpoint is 1800 psig, so an SI would not be warranted.

B. Incorrect but plausible. 15% VCT level is a criteria associated with a procedurally driven direction for a reactor trip, however it is associated with OS1227.02, “Steam Generator Tube

Leak" vice OS1201.02. Additionally, low VCT level would not warrant an SI.

C. **Correct.** Per OS1201.02, "RCS Leak" OAS page:

IF Plant in mode 1,2, or 3 with SI Accumulators aligned for injection AND PZR level can NOT be maintained greater than 7% using normal charging lineup, THEN perform the following:

1) Trip reactor
2) WHEN reactor trip is verified, THEN actuate SI
3) Go to E-0, REACTOR TRIP OR SAFETY INJECTION.

D. Incorrect but plausible. There is a procedurally driven action to manually trip the reactor and actuate SI if PZR level cannot be maintained >7% with two charging pumps, however the action is associated with OS1227.02, "Steam Generator Tube Leak" vice OS1201.02.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1201.02, RCS Leak (Rev 17)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1180I07RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	**10**		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q4	Tier #	1	
	Group #	1	
	K/A #	**011** Large Break LOCA **EK1** Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA : **EK1.01** Natural circulation and cooling, including reflux boiling	
	Importance Rating	4.1	

Proposed Question:

Given the following:

- A LOCA has occurred approximately 20 minutes ago.
- All equipment is operating as designed.
- RCS pressure is 200 psig and slowly lowering.
- Steam generator pressures are approximately 800 psig and slowly lowering.

Which ONE of the following describes the heat removal mechanism(s) currently occurring, and the operator action(s) that will be effective to enhance core cooling?

A. Break flow only; ensure adequate ECCS flow for current RCS pressure ONLY.

B. Break flow only; ensure RHR flow is rising as RCS pressure lowers, ensure adequate EFW flow exists, and operate ASDVs as necessary for plant cooldown.

C. Break flow and reflux boiling; ensure adequate ECCS flow for current RCS pressure ONLY.

D. Break flow and reflux boiling; ensure RHR flow is rising as RCS pressure lowers, ensure adequate EFW flow exists, and operate ASDVs as necessary for plant cooldown.

Proposed Answer: A

Explanation (Optional):

A. **Correct**. For large break LOCAs, break flow is the heat removal mechanism. At this point in the event, SGs are a heat source, and steaming them will provide no benefit for RCS cooldown until SG pressure is below RCS pressure.

B. Incorrect. Steaming SGs will enhance reflux cooling, but SGs must be a heat sink for this to occur.

C. Incorrect. Plausible because the action is correct, and applicant may misunderstand reflux cooling mechanism.

D. Incorrect. Plausible because the action is correct if reflux cooling were occurring, and applicant may misunderstand reflux cooling mechanism.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	UFSAR 15.6.5.2.3.1 Large Break LOCA Reference Split Break Transient Description (Rev 16)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1413I03RO			(As available)
Question Source:	Bank #	X	95903	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Wolf Creek		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	14	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q5	Tier #	1	
	Group #	1	
	K/A #	**015/017** Reactor Coolant Pump (RCP) Malfunctions **AK2**. Knowledge of the interrelations between the Reactor Coolant Pump Malfunctions (Loss of RC Flow) and the following: **AK2.07** RCP seals	
	Importance Rating	2.9	

Proposed Question:

Which one of the following explains the reason for closing the seal return valve after securing a Reactor Coolant Pump with a high #1 Seal Leakoff flow?

A. Prevent overflowing the RCP standpipe.

B. Establish a pressure boundary at the #2 seal.

C. Minimize heat load on seal return heat exchanger.

D. Prevent flow damage to the Thermal Barrier Heat Exchanger.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. It is plausible that if the #1 seal had high leakoff flow then the #2 seal would encounter a higher pressure drop, with fluid then backflowing through the #3 seal to the standpipe.

B. **Correct.** Backpressure on the #1 seal must be maintained in order to preserve the pressure breakdown capabilities of the #2 seal.

C. Incorrect but plausible. It is true that high #1 leakoff flow equates to an increase in fluid flow through the seal return heat exchanger, however this is not the reason for closing the seal return valve.

D. Incorrect but plausible. The Thermal Barrier Heat Exchanger is associated with the #1 seal, however, increased flow through the Thermal Barrier Heat Exchanger would be associated with a loss of seal injection vice high #1 seal leakoff flow.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	RCS Detailed System Text Sect 5.4.4 (Rev 9) OS1201.01, RCP Malfunction (Rev 18)				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	L1181I03RO, L8021I27RO			(As available)	
Question Source:	Bank #	X	101966		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2003 Robinson			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	7			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q6	Tier #	1	
	Group #	1	
	K/A #	**022** Loss of Reactor Coolant Makeup **AA2**. Ability to determine and interpret the following as they apply to | the Loss of Reactor Coolant Makeup: **AA2.04** How long PZR level can be maintained within limits	
	Importance Rating	2.9	

Proposed Question:

Plant conditions:

- 60% power
- All charging flow was lost.
- The crew entered OS1202.02, "Charging System Failure".
- Letdown has been isolated and charging restoration is being investigated.
- The PSO reports PZR level is lowering at a rate of 1% every five (5) minutes.
- PZR Level was 4% below reference when letdown was isolated.

If charging flow is NOT restored, which ONE of the following is the longest time that PZR heater operation can be maintained?

A. 40 minutes

B. 85 minutes

C. 125 minutes

D. 145 minutes

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. If the student misinterpreted or misapplied the data from the question stem then they could possibly calculate the time from minimum programmed level to the heater

cutout setpoint, which would be (25%-17%)(5min/%)=40 minutes

B. Incorrect but plausible. If the student misinterpreted or misapplied the data from the question stem they could possibly calculate the time from the initial level to the minimum programmed level, which would be (42%-25%)(5min/%)=85 minutes

C. **Correct.** At 60% power pressurizer level is at 60% of its full programmed range. The programmed range is 25%(@ 0%power/557°F) to 60%(@ 100% power/589°F). 60% of programmed range is 25% + (.6)([60%-25%]) = 46%. Initial level was 46%-4%=42%. The pressurizer heater cutout setpoint is 17%. The time for pressurizer level to drop to the heater cutout setpoint is (42%-17%)(5min/%)=125 minutes

D. Incorrect but plausible. If the student misinterpreted or misapplied the data from the question stem they could possibly calculate the time from the 60% power programmed level to the heater cutout setpoint, which would be (46%-17%)(5min/%)=145 minutes

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OX1401.02, RCS Steady State Leak Rate Calc (Rev 08)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #	X	101967	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2012 Beaver Valley		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q7	Tier #	1	
	Group #	1	
	K/A #	**025** Loss of Residual Heat Removal System (RHRS) **AA1**. Ability to operate and / or monitor the following as they apply to the Loss of Residual Heat Removal System: **AA1.08** RHR cooler inlet and outlet temperature indicators	
	Importance Rating	2.9*	

Proposed Question:

Plant conditions:

- MODE 4.
- "A"RHR in service.
- RCS temperature is 340°F and stable.
- "A" RHR HX inlet and outlet temperatures currently indicate as follows:
- Inlet = 340°F; Outlet = 310°F
- An electrical fault in the MOV circuit for CC-V-145, "A RHR HX PCCW Isolation valve" causes it to close from 100% to 75% open.

With no operator action how does the "A" RHR temperature control valve and HX Outlet temperature respond, if at all?

	Temperature Control Valve	HX Outlet Temp
A.	Opens	No change
B.	No change	Increases
C.	Closes	No Change
D.	Opens	Increases

Proposed Answer:	B	

Explanation (Optional):

A. Incorrect but plausible. The RHR system flow control valve and pump recirc valves do function automatically based on system parameter input. It is plausible that the student could incorrectly believe that the temperature control valve would reposition based on system parameter input. Given that misconception, it would be plausible that the temperature control valve would reposition such that it maintains a constant heat exchanger outlet temperature.

B. **Correct.** The RHR temperature control valve is an air operated valve that's position is controlled by a potentiometer that the control room operator manipulates. The valve receives no positioning signal from system parameters. The valve would not reposition without operator interface. Since the heat exchangers cooling water isolation valve repositions from full open to 75% open, RHR cooling capacity is reduced. The result would be an increase in heat exchanger outlet temperature.

C. Incorrect but plausible. The RHR system flow control valve and pump recirc valves do function automatically based on system parameter input. It is plausible that the student could incorrectly believe that the temperature control valve would reposition based on system parameter input. Additionally, if the student incorrectly believed that the RHR temperature control valve was configured such that it controlled heat exchanger bypass flow (as is the case with the PCCW system temperature control valve) then it is plausible that the temperature control valve would throttle closed so as to direct more RHR process flow through the heat exchanger, thus resulting in no change to the heat exchanger outlet temperature.

D. Incorrect but plausible. It is true that the heat exchanger outlet temperature would increase, however the temperature control valve would not reposition. As is the case with distractor "A", it is plausible that the student could incorrectly believe that the temperature control valve would reposition based on system parameter input. Given that misconception, it would be plausible that the temperature control valve would reposition in an attempt to stop the increasing heat exchanger outlet temperature.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-506651, RH Heat EX E-9a (Rev 10) 1-NHY-506652, RH Heat EX E-9a By-pass Vlv (Rev 10)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8033I07RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure

to provide the information will necessitate a detailed review of every question.)			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q8	Tier #	1	
	Group #	1	
	K/A #	**026** Loss of Component Cooling Water (CCW) **AA1**. Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: **AA1. 07** Flow rates to the components and systems that are serviced by the CCWS; interactions among the components	
	Importance Rating	2.9	

Proposed Question:

Which of the following describes what happens to the Thermal Barrier Cooling Water System when a high thermal barrier heat exchanger outlet flow is sensed?

A. Thermal Barrier Cooling Water Pump is automatically tripped.

B. Thermal Barrier PCCW Containment isolation valves automatically close.

C. Affected RCP thermal barrier heat exchanger outlet Motor Operated Valve will close once the NSO closes the breaker for the valve.

D. Affected RCP thermal barrier heat exchanger outlet Motor Operated Valve will automatically close and automatic isolation of remaining thermal barrier heat exchangers is blocked.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. The thermal barrier cooling water pump circuitry does have a flow related interlock, however it is a start permissive vice a trip interlock. It is plausible that the student may incorrectly believe that there is a pump high flow trip interlock vice a start permissive.

B. Incorrect but plausible. The PCCW system containment isolation valves will close based on automatic isolation signals, however the PCCW thermal barrier loop does not have automatic containment isolation capability.

C. **Correct.** The RCP thermal barrier heat exchangers are designed to automatically close on high flow, however, per Seabrook Station operating procedures these valves are normally de-energized. The reason for this alignment is to prevent a cascading of high flow conditions and resulting isolation of flow to the remaining three RCP thermal barrier heat exchangers. In the event that there is a high flow condition through a thermal barrier heat exchanger, Abnormal Operating Procedure OS1212.01, "PCCW System Malfunction" will direct isolation of the applicable heat exchanger by energizing the associated heat exchanger isolation valve.

D. Incorrect but plausible. A Seabrook Station specific concern for cascading high flow conditions (see description for answer "C") warrants the ability to prevent isolation of the remaining three RCP thermal barrier heat exchangers. It is plausible that there would be a high flow block feature for the remaining three isolation valves, as there is such a scheme for the plants EFW system flow control valves, however this is not the case for the Thermal Barrier System. The isolation valves are de-energized, and in the event that there is a high flow condition Abnormal Operating Procedure OS1212.01, "PCCW System Malfunction" will direct isolation of the applicable heat exchanger by energizing the associated heat exchanger isolation valve.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1212.01, PCCW System Malfunction (Rev 13) 1-NHY-506203, CC – RC pump Thermal Barrier Control Loop Diagram (Rev 5) 1-NHY-503274, CC- Thermal Barrier Isol Valve Logic Diagram (Rev 7)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1445I05RO			(As available)
Question Source:	Bank #	X	101968	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:
TEB27654

Examination Outline Cross-reference:	Level	RO	SRO
Q9	Tier #	1	
	Group #	1	
	K/A #	**027** Pressurizer Pressure Control System (PZR PCS) Malfunction **AK2**. Knowledge of the interrelations between the Pressurizer Pressure Control Malfunctions and the following: **AK2.03** Controllers and positioners	
	Importance Rating	2.6	

Proposed Question:

Plant conditions:

- MODE 4
- RCS pressure is 350 psig
- All wide range temperatures are 280°F

Predict the plant response to RCS wide range pressure transmitter PT-405 failing high with NO operator action?

A. Only "A" PORV would open.

B. Only "B" PORV would open.

C. Both PORVs would open.

D. Neither PORV would open.

Proposed Answer:	A

Explanation (Optional):

A. **Correct**. Per LTOP circuitry, both trains of LTOP are armed if all wide range temperature instruments are less than 290°F. The stem of the question states that all wide range temperatures are 280°F. Wide Range pressure channel PT-405 provides the pressure input for the “A” PORV opening signal. Since the “A” PORV is armed, a high failure of PT-405 would open the “A” PORV. The “B” PORV is armed, however it does not open as its associated pressure instrument is PT-403.

B. Incorrect but plausible. It is true that the “B” PORV would be armed, however its associated pressure instrument is PT-403. It is a common operator misconception that pressure channel PT-403 would be associated with the “A” PORV and pressure channel PT-405 would be

associated with the "B" PORV as this alpha-numeric association would align sequentially, as is the case with many pieces of train related equipment, for example, Pressurizer Pressure channels PT-455,456,457, and 458 are sequenced with trains A,B,C, and D signal and protective circuits.

C. Incorrect but plausible. It is true that both PORVs would be armed, however failure of a single pressure channel would not open both PORVs. It is plausible for the student to rationalize that the pressure input signals for the PORVs would utilize a ½ logic, as that may seem conservative with regard to defense against cold overpressure conditions.

D. Incorrect but plausible. It is true that both PORVs would be armed, however failure of a single pressure channel would only open its associated PORV. It is plausible for the student to rationalize that the pressure input signals for the PORVs would utilize a 2/2 logic, as that may seem conservative with regard to inadvertent PORV actuation and unnecessary depressurization.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509038, RCS Cold Overpressurization Control Block Diagram (rev5) OS1201.09, RCS WR Press or Temp Inst Failure Att A Simplified LTOP Drawing (rev 13)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L8027I07RO			(As available)
Question Source:	Bank #	X	94954	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2008 Salem		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q10	Tier #	1	
	Group #	1	
	K/A #	**029** Anticipated Transient Without Scram (ATWS) **EK2** Knowledge of the interrelations between the and the following an ATWS: **EK2.06** Breakers, relays, and disconnects	
	Importance Rating	2.9*	

Proposed Question:

Plant conditions:

- An ATWS has occurred.
- The crew has shut down the reactor using manual rod insertion and boration.
- One PORV stuck open on the initial pressure transient, resulting in Safety Injection actuation.
- The stuck open PORV has been ISOLATED.
- RCS pressure is 1900 psig and rising.

The crew has transitioned to ES-1.1, SI TERMINATION. When the Primary Board Operator attempts to reset SI, both trains reset but neither train blocks.

What is the cause of SI failing to block?

A. The timer in the Safety Injection Block/Reset logic has not timed out.
B. The initiating condition causing the SI actuation has not cleared.
C. The Reactor Trip Breakers are closed.
D. P-11 has not been blocked.

Proposed Answer:	C

Explanation (Optional):

Comment: There has historically been an operator misconception associated with the SI reset/block logic scheme, particularly with regard to the logic scheme input from a) a 60 second reset permissive timer, b) P-4 signal input, and c) associated automatic or manual SI signal input. The plausibility of distractors "A" and "B" are associated with this misconception.

A. Incorrect but plausible. It is true that there is a 60 second timer that must time-out, however given that the crew has transitioned to ES-1.1, "SI Termination", the timeframe would be beyond the 60 second requirement.

B. Incorrect but plausible. It is true that the SI input signal is associated with the reset/block logic, however it does not prevent the ability to reset the signal when initiation signal has reset.

C. **Correct.** In order to reset the SI signals a 60 second timer must time out and the associated train P-4 signal must be reset. The P-4 signal is based on reactor trip breaker position. The breakers must be opened in order to reset SI and the stem of the question indicates the reactor trip breakers are closed.

D. Incorrect but plausible. P-11 is associated with SI, however it allows for manual block of the low pressurizer pressure and low steamline pressure SI when P-11 resets below 1941 psig.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509048, Safeguards Actuation Signal W Functional Diagram (Rev 17)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8056I29RO			(As available)
Question Source:	Bank #	X	101969	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
TEB30076

Examination Outline Cross-reference:	Level	RO	SRO
Q11	Tier #	1	
	Group #	1	
	K/A #	**038** Steam Generator Tube Rupture (SGTR) **EA1** Ability to operate and monitor the following as they apply to a SGTR: **EA1.40** Adding boron, to raise its ppm to the required shutdown concentration	
	Importance Rating	4.0	

Proposed Question:

Plant conditions:

- SG tube rupture has occurred.
- E-3, "Steam Generator Tube Rupture" has been completed.
- ES-3.1, "Post – SGTR Cooldown Using Backfill" has been entered.
- All rods are fully inserted.
- Rapid boration is in progress.
- RCS boron concentration sample has been requested.
- The crew is determining if shutdown margin is adequate for cooldown to commence.

With regard to shutdown margin (SDM) can the cooldown begin at this time?

A. Yes. SDM is maintained with all rods fully inserted.
B. Yes. SDM is ensured by previous SI and rapid boration.
C. Yes. The cooldown rate limit in ES-3.1 ensures SDM is maintained.
D. No. RCS sampling and analysis have not been completed as required by ES-3.1.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect but plausible. All rods fully inserted does provide SDM in the event of a reactor trip, however given the conditions in the stem of the question SDM is maintained via ECCS injection and subsequent rapid boration to account for the RCS cooldown and backfill from the SG.

B. **Correct.** SDM is maintained via ECCS injection and subsequent rapid boration to account for the RCS cooldown and backfill from the SG.

C. Incorrect but plausible. It is plausible that the limit on cooldown rate could be associated with SDM, to allow for the concurrent boration to adequately increase RCS boron concentration. The cooldown rate is actually limited to ensure more uniform fluid temperatures and to minimize the possibility of voiding in the primary system during depressurization.

D. Incorrect but plausible. ES-3.1 does have a step to verify adequate Cold Shutdown Boron Concentration and borate if necessary, but it does not prevent proceeding with the cooldown. There is an EOP procedure that does have a stop in it until SDM is verified (ES-0.2).

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-3.1, Post SGTR Cooldown Using Backfill (Rev 28) ES-3.1 Background document (Rev 2)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1206I17RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**10**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q12	Tier #	1	
	Group #	1	
	K/A #	**054** Loss of Main Feedwater (MFW) **AK3**. Knowledge of the reasons for the following responses as they apply to the Loss of Main Feedwater (MFW): **AK3.01** Reactor and/or turbine trip, manual and automatic	
	Importance Rating	4.1	

Proposed Question:

Plant conditions:

- 14% power.
- The turbine is rolling up to 1800 rpm.
- "A" MFP is in service.
- "A" Condensate Pump is tagged out.
- "B" and "C" Condensate Pumps have just tripped.

Assuming CO pressure drops to 0 psig upon pumps tripping, which one of the following combinations will occur if no operator action is taken?

	MFP TRIPs:	REACTOR TRIPs on:	TURBINE TRIPs on:
A.	After 12 seconds	Lo-Lo S/G Level	AMSAC
B.	After 12 seconds	Turbine Trip	Reactor Trip
C.	After 6 seconds	Turbine Trip	AMSAC
D.	After 6 seconds	Lo-Lo S/G Level	Reactor Trip

Proposed Answer: D

Explanation (Optional):

A. Incorrect. MFPs trip when <220 psi suction pressure ("A" MFP has 6 second delay, "B" MFP has 12 second delay). AMSAC is not armed at 14% power.

B. Incorrect. MFPs trip when <220 psi suction pressure ("A" MFP has 6 second delay, "B" MFP has 12 second delay). Turbine trip less than 45% will not cause the Rx to trip. AMSAC is not armed at 14% power.

C. Incorrect. Turbine trip less than 45% will not cause the Rx to trip. AMSAC is not armed at 14% power.

D. **Correct.** MFPs trip when <220 psi suction pressure ("A" MFP has 6 second delay, "B" MFP has 12 second delay). With no feed to the S/Gs, the levels will drop and at 20% in 2/4 levels in one S/G a Rx Trip is generated which in turn generates a Turbine Trip

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503592, FW Pump Turbine Pump Suct Press Trip Logic (rev13) FP700094 sh 3 of 45, A MFP Global Variable Report (rev 0) FP700100 sh 3 of 44, B MFP Global Variable Report (rev 0)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8062I08RO			(As available)
Question Source:	Bank #	X	93986	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2008 Ginna		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q13	Tier #	1	
	Group #	1	
	K/A #	**056** Loss of Offsite Power **AK1**. Knowledge of the operational implications of the following concepts as they apply to Loss of Offsite Power: **AK1.01** Principle of cooling by natural convection	
	Importance Rating	3.7	

Proposed Question:

Why is it important to run the CRDM fans when performing a natural circulation cooldown?

A. Aids the operator in maintaining RCS subcooling.

B. Allows the operator to cooldown the RCS greater than 100°F/hr.

C. Provides the major heat removal mechanism for the vessel head area.

D. Ensures the heat generated by the CRDM units is not added to the RCS.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. Operating CRDM fans allows the operators to depressurize the RCS to a lower subcooling value than would be allowed without CRDM fans, however this is associated with cooling the vessel head to prevent void formation in the head vice maintaining RCS subcooling.

B. Incorrect but plausible. Running CRDM fans allows the operators to depressurize more aggressively, however it does not allow the operators to cool down at a faster rate.

C. **Correct.** Per the Westinghouse ES-0.2, "Natural Circulation Cooldown" background document, "operation of CRDM fans significantly aids in removing heat from the upper vessel head area".

D. Incorrect but plausible. The CRDM fans are operated to remove heat, however it is to cool down the vessel upper head region vice prevent adding heat to the head.

<table>
<tr><td>Technical Reference(s):
(Attach if not previously provided)
(including version/revision number)</td><td colspan="5">ES-0.2, Natural Circulation Cooldown (Rev 34)</td></tr>
<tr><td colspan="5">Proposed references to be provided to applicants during examination:</td><td>None</td></tr>
<tr><td>Learning Objective:</td><td colspan="4">L1225I14RO</td><td>(As available)</td></tr>
<tr><td>Question Source:</td><td>Bank #</td><td>X</td><td>101971</td><td colspan="2"></td></tr>
<tr><td></td><td>Modified Bank#</td><td></td><td></td><td colspan="2">(Note changes or attach Parent)</td></tr>
<tr><td></td><td>New</td><td></td><td></td><td colspan="2"></td></tr>
<tr><td>Question History:</td><td>Last NRC Exam</td><td colspan="4"></td></tr>
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<tr><td colspan="3">Comprehension or Analysis</td><td></td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="4">10</td></tr>
<tr><td>55.43</td><td colspan="4"></td></tr>
<tr><td colspan="6">Comments:
TEB20670</td></tr>
</table>

Examination Outline Cross-reference:	Level	RO	SRO
Q14	Tier #	1	
	Group #	1	
	K/A #	**057** Loss of Vital AC Electrical Instrument Bus **AA2**. Ability to determine and interpret the following as they apply to the Loss of Vital AC Instrument Bus: **AA2.16** Normal and abnormal PZR level for various modes of plant operation	
	Importance Rating	3.0	

Proposed Question:

Initial plant conditions:

- 50% power.
- Tave is 573°F and stable.
- PZR level is 42% and stable.

An event occurs and the PSO reports the following:

- PZR level is 44% and rising.

Which of the following events could be causing the PZR level response?

A. Loss of PP-1B.

B. Loop 1 NR Tcold fails low.

C. RC-LK-459, PZR Level Controller output fails low.

D. MS-PT-3001, "A" MS Pressure input to "A" ASDV fails high

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Per OS1247.01, "Loss of a 120 VAC Vital Instrument Panel" a loss of PP-1A or PP-1B will result in loss of letdown. Specifically, a loss of PP-1B will result in de-energization/low

failure of pressurizer level channel 460 and subsequent closure of RC-LCV-460, "Letdown Isolation Valve".

B. Incorrect but plausible. The pressurizer level control system has a ramped level setpoint from 0-100% power. Plant power level input to the control system is derived from RCS Avg Tavg. Narrow range Tcold values do input into the Avg Tavg calculation. A low failure of a Tcold instrument would cause the calculated Avg Tavg to decrease, thus lowering the pressurizer level setpoint. Actual pressurizer level would then be higher than setpoint, causing the charging flow control valve to throttle in the closed direction. This would cause pressurizer level to lower vice rise.

C. Incorrect but plausible. A failure of the RC-LK-459 output signal does affect charging flow, however if the output signal of RC-LK-459 fails low then charging flow control valve would throttle in the closed direction. This would cause pressurizer level to lower vice rise.

D. Incorrect but plausible. High failure of MS-PT-3001 would cause the "A" ASDV to open, which would affect RCS temperature and pressurizer level, however, a cooldown of the reactor coolant system would result in a lowering pressurizer level transient vice rising.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1247.01, Loss of a 120 VAC Vital Instrument Panel (PP1a, 1B, 1C or 1D) (Rev 17)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1186I08RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q15	Tier #	1	
	Group #	1	
	K/A #	**062** Loss of Nuclear Service Water **2.2.12** Knowledge of surveillance procedures.	
	Importance Rating	3.7	

Proposed Question:

OX1416.09, "Monthly Cooling Tower Portable Pump Operability Surveillance" is in progress.

Which of the following sets of conditions meets the requirements for the design operational readiness state of this pump?

A. Pump located in SW pump house. Fuel tank empty. Thirty (30) sections of hose covered with fire blanket.

B. Pump located adjacent to cooling tower. Fuel tank empty. Thirty (30) sections of hose covered with fire blanket.

C. Pump located in SW pump house. Fuel tank topped off. Thirty (30) sections of hose connected to pump and on trailer.

D. Pump located adjacent to cooling tower. Fuel tank topped off. Thirty (30) sections of hose connected to pump and on trailer.

Proposed Answer:	A

Explanation (Optional):

KA Match Justification: The KA is associated with category 062, Loss of Nuclear Service Water. The assigned KA is general KA 2.2.12, Knowledge of Surveillance Procedures. Procedure OS1247.02, "Degraded Ultimate Heat Sink", includes steps for making up inventory to the Cooling Tower. If the normal means of makeup is unavailable, then the procedure directs the operators to makeup to the tower with the portable tower makeup pump. This direction is included in steps 6j RNO, 7c RNO, and 8c RNO. A designated surveillance procedure, OX1416.09, "Monthly Cooling Tower Portable Pump Operability Surveillance ensures that the portable cooling tower pumps is readily available for this specific purpose.

A. **Correct.** Per procedure OX1416.09, "Monthly Cooling Tower Portable Pump Operability Surveillance", the portable makeup pump is verified to be in a state of readiness. Per table 1 of the procedure acceptance criteria for "yellow discharge hose" is "at least 30 rolls". Per table 2, the acceptance criteria requires that the suction hose, discharge hoses, and suction strainer are "covered with fire blankets". Per table 3, the acceptance criteria requires that the fuel oil tank be "empty". Additionally, location of the pump in the SW pumphouse meets the requirements of being in a seismic building.

B. Incorrect but plausible. It is true that the pump fuel tank must be empty and there must be 30 sections of hose covered with a fire blanket, however the pump must be located inside a seismic building.

C. Incorrect but plausible. It is true that location of the pump in the SW pumphouse meets the requirements of being in a seismic building and that there must be 30 sections of hose, however the fuel tank must be empty and the hose is not required to be connected to the pump.

D. Incorrect but plausible. It is true that there must be 30 sections of hose, however the pump must be located inside a seismic building. Additionally, the hose is not required to be connected to the pump.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.4.7 Service Water System/Ultimate Heat Sink Basis (Rev 118) OX1416.09, Monthly Cooling Tower Portable Pump Operability Surveillance (Rev 9)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8037I15RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	**10**	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q16	Tier #	1	
	Group #	1	
	K/A #	**065** Loss of Instrument Air **2.4.45** Ability to prioritize and interpret the significance of each annunciator or alarm.	
	Importance Rating	4.1	

Proposed Question:

Plant conditions:

- A loss of service air has occurred.
- ON1242.01, "Loss of Instrument Air" is being processed.
- One air compressor has been started and IA pressure is 90 psig and lowering.

Which of the following alarms if received requires the reactor to be tripped?

A. B8347, "FEED REG VLV AVG POSITION >90% OPEN".

B. MM-UA-52 (C-8), "MASTER PRESS CTLR OUTPUT HI".

C. MM-UA-50 (F-3), "PCCW RCP COOLERS FLOW LO".

D. B6942, "PCCW TRN A SUPPLY HDR TEMP LO-LO".

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. The feedwater reg. valves are air operated valves, however they fail closed on a loss of instrument air.

B. Incorrect but plausible. CS-LCV-121, "Charging Flow Control" does fail open on a loss of instrument air. Additionally CS-V-145, "Regen Heat Exchanger Outlet" does fail closed on a loss of instrument air, so it is conceivable that a loss of instrument air would result in a pressurizer pressure transient due a significant flow imbalance between letdown and charging. The flow imbalance would cause an increase in pressurizer level and pressure, however the "Master Press CTLR Output Hi" alarm is not a condition that necessarily warrants a reactor trip.

C. **Correct.** Per ON1242.01, "Loss of Instrument Air", OAS page, the RCPs must be tripped within 10 minutes of losing PCCW flow to containment. The "PCCW RCP Coolers Flow Low" alarm is indicative of a loss of PCCW flow to containment.

D. Incorrect but plausible. The PCCW heat exchanger outlet valves fails open and the heat exchanger bypass valves fail closed, which would result in a significant decrease in PCCW system temperature, however the "PCCW TRN A Supply HDR Temp Lo-Lo" alarm is not a condition that necessarily warrants a reactor trip.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1242.01, Loss of Instrument Air (Rev 13)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1194I03RO				(As available)
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q17	Tier #	1	
	Group #	1	
	K/A #	**W/E11** Loss of Emergency Coolant Recirculation **2.4.3** Ability to identify post-accident instrumentation.	
	Importance Rating	3.7	

Proposed Question:

Plant conditions:

- A Large Break LOCA coincident with a Loss of Off-site power has occurred.
- A fault on Bus E-5 prevents restoration of "A" train ECCS.
- 20 minutes later the crew has entered ECA-1.1, "Loss of Emergency Coolant Recirculation" due to CBS-V-14 failing to open.
- The PSO is directed to check RWST level greater the 80,000 gallons.

Which ONE (1) of the following identifies instrumentation used for monitoring and operating the Containment Building Spray System with these conditions?

A. A0912, RWST LEVEL

B. CBS-LI-2381, RWST NR Level

C. CBS-LI-2380, RWST WR Level

D. CBS LI-930, RWST Level (CH I)

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. Plausible that MPCS computer indications give better resolution for RWST level being at 80,000 gallons. Incorrect as 15 minutes after bus 5 loses power the MPCS is unavailable.

B. Incorrect but plausible. Plausible that NR RWST level gives more accurate indication. Incorrect as the NR level meter only reads 420,000 to 490,000 gallons.

C. **Correct.** WR level instrument is a PAM instrument and will be the only indication under the

above conditions to accurately display RWST level.

D. Incorrect but plausible. Plausible that protection CHI instrument that is used for auto swap over logic from RWST to containment sumps would accurately indicate low level of RWST. Incorrect as the protection channel RWST level instruments have no level indications, they are used for bistable inputs only

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-506170, CBS SAT and RWST Control Loop Diagram (Rev17)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	7		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q18	Tier #	1	
	Group #	1	
	K/A #	**077** Generator Voltage and Electric Grid Disturbances **2.4.4** Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	
	Importance Rating	4.5	

Proposed Question:

Plant conditions:

- 100% power.
- At 0900, voltages on all three offsite lines degraded to 340 KV due to a disturbance on the grid.
- Bus E5 and E6 Voltages have degraded to 3900 Volts, stable.
- OS1246.02, "Degraded Vital AC Power (Plant Operating)" is being implemented.
- At 0915, ISO New England notifies the Control Room that grid voltage cannot be restored until 1030, at the earliest.

What action is required?

A. Transfer Bus E5 and E6 to EDGs. When grid voltage is restored, restore off site power to Bus E5 and E6.

B. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". Following immediate actions, transfer Bus E5 or E6 to EDG.

C. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". Following immediate actions, transfer Bus E5 and E6 to EDGs.

D. Transfer Bus E5 or E6 to EDG. When grid voltage is restored, restore off site power to Bus E5 or E6.

Proposed Answer:	C

Explanation (Optional):

A. Incorrect but plausible. It is true that OS1246.02, "Degraded Vital AC Power (Plant Operating), includes a strategy for transferring an emergency bus to the EDG's without the need for a reactor trip, however step 15e checks to see if both AC emergency busses are impacted by the degraded voltage condition. If only one emergency bus is affected then the procedure step RNO action directs that the one affected emergency bus (with voltage less than 3933 volts for greater than 15 minutes) be transferred to its associated emergency diesel generator. It is plausible that the student would misapply the one bus affected versus both busses affected strategies.

B. Incorrect but plausible. Per OS1246.02, "Degraded Vital AC Power (Plant Operating), step 16a, if both AC emergency busses (Bus E5 and E6) voltages have been less that 3933 volts for greater than 15 minutes then step 16b directs tripping the reactor, entering procedure E-0, performing E-0 immediate actions, however the procedure step then directs then transferring busses E5 and E6 to the emergency diesel generators vice transferring a single bus.

C. **Correct.** Per OS1246.02, "Degraded Vital AC Power (Plant Operating), step 16a, if both AC emergency busses (Bus E5 and E6) voltages have been less than 3933 volts for greater than 15 minutes then step 16b directs tripping the reactor, entering procedure E-0, performing E-0 immediate actions, and then transferring busses E5 and E6 to the emergency diesel generators.

D. Incorrect but plausible. It is true that OS1246.02, "Degraded Vital AC Power (Plant Operating), includes a strategy for transferring an emergency bus to the EDG's without the need for a reactor trip, however step 15e checks to see if both AC emergency busses are impacted by the degraded voltage condition. If only one emergency bus is affected then the procedure step RNO action directs that the one affected emergency bus (with voltage less than 3933 volts for greater than 15 minutes) be transferred to its associated emergency diesel generator. It is plausible that the student would misapply the one bus affected versus both busses affected strategies.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1246.02, Degraded Vital AC Power (Plant Operating) (Rev 15)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1199I11RO			(As available)
Question Source:	Bank #	X	101972	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55	55.41	**10**	

	55.43	
Comments: TEB31598		

Examination Outline Cross-reference:	Level	RO	SRO
Q19	Tier #	1	
	Group #	2	
	K/A #	**028** Pressurizer (PZR) Level Control Malfunction **AK2**. Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the following: **AK2.02** Sensors and detectors	
	Importance Rating	2.6	

Proposed Question:

Plant Conditions:

- 100% power.
- All control systems are in their normal alignment.
- A PZR level transmitter sensing line leak occurs.
- Charging flow increases to maximum.
- Letdown automatically isolates.

Which PZR level transmitter is affected and which sensing line is leaking?

	Transmitter	Sensing line
A.	459	Variable
B.	459	Reference
C.	460	Variable
D.	460	Reference

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Normal alignment for the pressurizer level control system utilizes level channel 459 as the primary channel and 460 as the backup channel. The primary channel supplies an input signal to the pressurizer level controller, which then provides an input signal to the controller for CS-FCV-121, "Charging Flow Control Valve". Both the primary and backup channels input to separate bistables that cause a letdown isolation signal and pressurizer heater cutout signal on low pressurizer level (<17%). If the variable leg of the 459 level transmitter had a leak, then the

instrument would sense pressurizer level as being lower than actual level. This condition would result in an increase in charging flow and a letdown isolation signal.

B. Incorrect but plausible. It is true that level channel 459 is the affected channel, however a leak in the transmitters reference leg would result in that channel reading higher than actual level. If the student had a misconception with regard to the functional principles of a Δp type level transmitter then they could conclude that the leak were associated with the reference leg.

C. Incorrect but plausible. It is true that if the variable leg of the 460 level transmitter had a leak, then the instrument would sense pressurizer level as being lower than actual level. This condition would result in a letdown isolation signal, however that would cause pressurizer level to increase which in turn would cause charging flow to decrease. If the student had a system knowledge gap with regard to the functions supplied by the backup channel then they could conclude that the affected transmitter is 460.

D. Incorrect but plausible. If the student had a system knowledge gap with regard to the functions supplied by the backup channel then they could conclude that the affected transmitter is 460. Additionally, if the student had a misconception with regard to the functional principles of a Δp type level transmitter then they could conclude that the leak were associated with the reference leg.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509051, RC PRZR Press & Lvl CTL W Functional Diagram (Rev 7)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8027I14RO, L1182I01RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	101992	(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55	55.41	7		

	55.43	

Comments:

TEB14286

101992 prior to significant modification.

Plant Conditions:

- 100% power.
- Pressurizer level control selector switch is in the 459/460 position.
- A Pressurizer level channel failure occurs.
- Charging flow increases to maximum.
- Letdown automatically isolates.
- Pressurizer heaters trip.

Which Pressurizer level channel has failed and what is the direction of the failure?

	Channel	Direction
A.	459	HIGH
B.	459	LOW
C.	460	HIGH
D.	460	LOW

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Q20	Tier #	1	
	Group #	1	
	K/A #	**037** Steam Generator (S/G) Tube Leak **AA2**. Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: **AA2.10** Tech-Spec limits for RCS leakage	
	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- RCS leakage into "B" SG is 0.15 gpm.
- No leakage is detectable into the other SGs.
- RCS unidentified leakage is 0.7 gpm.
- RCS identified leakage, other than the SG leakage is 4.0 gpm.
- Pressure isolation valve leakage is 0.1 gpm.

Which RCS leakage LCO has been exceeded?

A. Identified leakage.
B. Unidentified leakage.
C. Steam generator tube leakage.
D. Pressure isolation valve leakage.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. Per Tech. Spec. section1.0, “Definitions”, RCS Identified Leakage includes primary to secondary leakage. Given the data in the question stem Identified Leakage is 4.15 gpm (4.0 gpm + .15 gpm) , which does not exceed Tech Spec 3.4.6.2, “Reactor Coolant

System Leakage", Identified Leakage limit of 10 gpm. It is plausible for the student to confuse the Identified Leakage limit with the Unidentified Leakage limit, which is 1 gpm.

B. Incorrect but plausible. It is plausible for the student to misinterpret primary to secondary leakage as pressure boundary leakage, as the steam generator tubes are a physical RCS pressure boundary. Additionally, it is plausible for the student to confuse the Unidentified Leakage limit with the Pressure Boundary leakage limit, which is limited to "No Pressure Boundary Leakage".

C. **Correct.** Per Tech Spec 3.4.6.2, "Reactor Coolant System Leakage", primary to secondary leakage through any one steam generator is limited to 150 gallons per day. The stem of the question states that there is .15 gpm leakage to the "B" steam generator. (.15 gal/min)(60 min/hr)(24 hr/day) = 216 gallons per day.

D. Incorrect but plausible. It is plausible for the student to misinterpret leakage through a pressure isolation valve as being pressure boundary leakage, as valves interfacing with the reactor coolant system are a physical boundary.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.4.6.2, RCS Operational Leakage (Rev 118)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1190I10RO			(As available)
Question Source:	Bank #	X	101973	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	

10 CFR Part 55 Content:	55.41	(CFR: 43.5 / 45.13)
	55.43	

Comments:
TEB28074

Examination Outline Cross-reference:	Level	RO	SRO
Q21	Tier #	1	
	Group #	1	
	K/A #	**051** Loss of Condenser Vacuum **AK3.** Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: **AK3.01** Loss of steam dump capability upon loss of condenser vacuum	
	Importance Rating	2.8*	

Proposed Question:

The following conditions exist:

- Circulating Water Pumps 39A & C have tripped which resulted in the crew tripping the reactor from full power.
- Steam Header Pressure is 1030 psig
- Tavg is 550°F
- Condenser Vacuum is 19.5"Hg(vac)
- Steam dumps are closed.

Which of the following correctly describes the operation of the steam dump system?

A. The steam dumps will open as soon as Tavg is greater than or equal to 559°F.

B. The steam dumps will not open because C-9 permissive requirements are not met.

C. The steam dumps will open when the operator bypass the Lo-Lo T_{avg} (P-12) interlock.

D. The steam dumps will not open until the operators place the steam dumps in the STEAM PRESSURE MODE.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. The C-9 permissive signal requires at least 1 CW pump breaker closed and condenser vacuum >25". The student could incorrectly assume that the C-9 permissive is met based on one CW pump breaker being closed.

B. **Correct**. The steam dump permissive circuit C-9 requires condenser vacuum to be >25". Given the conditions in the stem of the question, the steam dumps are disabled.

C. Incorrect but plausible. This distractor pertains to the Lo-Lo Tavg (P-12) interlock, which is the Tavg value given in the question stem. The student may choose this distractor if they focus on the Tavg/P-12 association and either fail to include the C-9 signal in their analysis, or misinterprets the C-9 signal (as described in answer "A"").

D. Incorrect but plausible. Placing the steam dumps in the "Steam Pressure Mode" will "arm" the steam dumps, however the C-9 "permissive" signal is not met. If the student misinterprets between the steam dump "arming" and "permissive" signals then they would incorrectly assume that placing the dumps in the "Steam Pressure Mode" would re-enable them.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Steam Dump Detailed System Text pg 17 (Rev 5) 1-NHY-509050, MS Dump Control (Rev 5)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8047I13RO, 15RO			(As available)
Question Source:	Bank #	X	TEB16326	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q22	Tier #	1	
	Group #	1	
	K/A #	**060** Accidental Gaseous Radwaste Release **AK3**. Knowledge of the reasons for the following responses as they apply to the Accidental Gaseous Radwaste: **AK3.03** Actions contained in EOP for accidental gaseous-waste release	
	Importance Rating	3.8	

Proposed Question:

A reactor trip and Safety Injection has occurred due to a large break LOCA.

The PSO is performing E-0, "Reactor Trip or Safety Injection" Attachment A, "ESF Actuation Verification".

What alignment is verified FIRST and why?

A. Feedwater Isolation. Prevent excessive RCS cooldown.

B. MS-V-129 Open. Maintain adequate heat sink for decay heat removal.

C. RCS Isolation. Prevent additional loss of RCS inventory and/or pressure.

D. Containment Isolation Phase A. Prevent release of radioactive materials from containment.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. Verification of "Feedwater Isolation" is important in preventing an uncontrolled filling of steam generators and associated RCS cooldown, which could aggravate the transient. Feedwater Isolation is checked in Attachment A, however it is not checked before "Containment Isolation Phase A".

B. Incorrect but plausible. Verification that MS-V-129 is OPEN is important for maintaining a secondary heat sink. MS-V-129 is checked open in Attachment A, however it is not checked before "Containment Isolation Phase A".

C. Incorrect but plausible. Verification that the RCS is isolated (letdown valves closed,

PORV's closed, and spray valves closed) is important in mitigating an uncontrolled RCS depressurization, loss of inventory, or potential steam space LOCA. RCS isolation verification is performed in E-0, however the step is contained in the main body of the procedure vice Attachment A.

D. **Correct**. Per E-0, "Reactor Trip or Safety Injection", Attachment A, "ESF Actuation Verification", "Containment Isolation Phase A" is the first alignment checked.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection, Attachment A, ESF Actuation Verification (Rev 50) E-0 Background document for step 6 (Rev 2)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1202I02RO				(As available)
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q23	Tier #	1	
	Group #	2	
	K/A #	**061** Area Radiation Monitoring (ARM) System Alarms **AK2**. Knowledge of the interrelations between the Area Radiation Monitoring (ARM) System Alarms and the following: **AK2.01** Detectors at each ARM system location	
	Importance Rating	2.5*	

Proposed Question:

Radiography is being performed on MS-V-393, "SG A MS Supply to Turbine Driven EFW pump".

The RP supervisor responsible for the evolution contacts the Control Room to inform the operators of RMS alarms which would be expected to be received during the performance of the radiography.

Which of the following Radiation Monitors would be included in the report by the RP supervisor?

A. RM-6568-1, "Containment Exhaust Radiation Monitor.

B. RM-6510-1, "A SG Blowdown Radiation Monitor".

C. RM-6482-2, "C MS Line Radiation Monitor".

D. RM-6481-2, "D MS Line Radiation Monitor".

Proposed Answer: D

Explanation (Optional):

A. Incorrect. CEVA radiation monitor is on the 84 ft of the PAB. Plausible as the CEVA is adjacent to the west pipe chase.

B. Incorrect. A SG Blowdown radiation monitor is on the 53 ft of the PAB. Plausible as all 4 SG blowdown lines penetrate containment in the West pipe chase and their isolation valves are located there.

C. Incorrect. MS line C radiation detector is in the East pipe chase. Plausible that the C steam line is in same pipe chase as the A steam line.

D. **Correct.** MS-V-393 is in the west pipechase middle level. MS line A and D radiation detectors

are in proximity to this valve.					
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	RM Detailed System text, Table 3.1 pg 29 (Rev 9)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:					(As available)
Question Source:	Bank #	X	100934		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2012 Point Beach			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41	7			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q24	Tier #	1	
	Group #	2	
	K/A #	**067** Plant fire on site **2.4.35** Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	
	Importance Rating	3.8	

Proposed Question:

Plant Conditions:

- A fire is in progress in the cable spreading room.
- OS1200.00, "Response to Fire or Fire Alarm Actuation" is in progress.
- A CRO has been sent to the "B" Essential Switchgear Room and is performing OS1200.02B, "Remote Safe Shutdown Control-Train B".

What direction should be given to the RSS NSO?

A. Go to the Boric Acid Tank Room; perform a manual rapid boration per US direction.

B. Go to DG 1B local control panel; obtain a copy of OS1200.02B from DG 1B procedure satellite, and man the sound powered phones.

C. Go to the SEPS Diesels; obtain a copy of ON1061.01, "Operation of SEPS, and man the sound powered phones.

D. Go to the EFW Pump house; obtain a copy of OS1036.03, "Resetting The Steam Driven EFW Pump Trip Valve, and man the sound powered phones.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. Local manual rapid boration is a local "auxiliary operator task during an emergency", however it is not associated with OS1200.00 or OS1200.02B. The task is associated with a required local boration in the event that boration capabilities are lost from the control room.

B. **Correct**. Per OS1200.00, "Response to Fire or Fire Alarm Actuation", step 2 RNO, if there is a fire in the cable spreading room the RSS NSO is directed to 1) go to DG 1B local control panel, obtain a copy of OS1200.02B from the DG 1B procedure satellite panel, and man the sound

powered phones on the DG 1B local control panel. Additionally, per OP 10.8, "Nuclear Systems Operator Watchstanding Practice", section 4.1.2, "NSO Remote Safe Shutdown Responsibilities, states that the RSS NSO (Secondary Nuclear System Operator) proceeds to the Train B emergency diesel room, mans the sound powered phones, and awaits direction from the US.

C. Incorrect but plausible. A RSS shutdown scenario does involve actions associated with the EDG's but not with the SEPS diesels. Additionally, there are NSO defined responsibilities associated with SEPS, however they are associated with the Primary NSO's response to an SI and/or LOP event.

D. Incorrect but plausible. There are NSO responsibilities associated with the Emergency Feedwater Pumps, however they are associated with the Roving NSO's response to a Reactor Trip/SI.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1200.00, Response to Fire Alarm Actuation (Rev 22) OS1200.02B, Remote Safe Shutdown Control – Train B (Rev 18) OP10.8, Nuclear Systems Operator Watch standing Practice (Rev 14)

Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	**10**	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q25	Tier #	1	
	Group #	2	
	K/A #	**W/E13** Steam Generator Overpressure **2.4.1** Knowledge of EOP entry conditions and immediate action steps.	
	Importance Rating	4.6	

Proposed Question:

Given the following conditions:

- Reactor trip and Safety Injection.
- The crew is performing the actions of E-1, "Loss of Reactor or Secondary Coolant".
- Intermediate Range SUR is -0.3 DPM.
- Intermediate Range NI level is 0.1% and decreasing.
- S/G NR levels are "A" 30%; "B" 25%; "C" 85%; "D" 30%.
- S/G pressures are "A" 1040 psig; "B" 1040 psig; "C"1240 psig; "D" 1125; psig.
- Total EFW flow is 0 GPM.
- RCS pressure is 1650 psig and stable.
- Core Exit Thermocouple Temperature is 565°F and stable.
- No RCPs are operating.
- RVLIS Full Range Level is 45% and stable.
- Containment Pressure is 3 psig and slowly rising.
- You have been assigned to monitor Critical Safety Function Status Trees.

Of the choices listed below which could you recommend the US implement at this time?

A. FR-H.2, "Response to Steam Generator Overpressure".
B. FR-C.2, "Response to Degraded Core Cooling".
C. FR-S.2, "Response to Loss of Core Shutdown".
D. FR-H.3, "Response to Steam Generator High Level".

Proposed Answer: A

Explanation (Optional):

A. **Correct**. Per the H CSF Status Tree, with any SG pressure > 1225 psig, FR-H.2 should be implemented.

B. Incorrect but plausible. This choice is plausible as most of the status tree criteria are met for FR-C.2, however RVLIS full range level would have to be less than 40%.

C. Incorrect but plausible. This choice is plausible as most of the status tree criteria are met for FR-S.2, however IR startup rate would have to be less negative than -.2 dpm

D. Incorrect but plausible. This choice is plausible as most of the status tree criteria are met for FR-H.3, however SG narrow range levels would have to be >90%.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	F-0.1, Subcriticality (S) (Rev 20) F-0.2, Core Cooling (C) (Rev 20) F-0.3, Heat Sink (H) (Rev21)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1211I05RO			(As available)
Question Source:	Bank #	X	100901	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2012 Point Beach		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**10**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q26	Tier #	1	
	Group #	1	
	K/A #	**W/E16** High Containment Radiation **EK1**. Knowledge of the operational implications of the following concepts as they apply to the (High Containment Radiation) **EK1.2** Normal, abnormal and emergency operating procedures associated with (High Containment Radiation).	
	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- A small break LOCA has occurred.
- The crew is progressing through ES-1.2, "Post LOCA Cooldown and Depressurization"
- All Critical Safety Functions are satisfied with the exception of Containment (Z) which has a yellow terminus based on post-accident radiation monitor reading > 10 R/HR.
- The US refers to FR-Z.3, "Response to High Containment Radiation Level".

Why is containment pressure verified less than 18 psig?

A. The "P" signal will prevent the containment recirculation filter fans from starting.

B. The "P" signal will prevent the containment recirculation filter system realignment to the Filter Mode.

C. The radioactive release associated with a containment pressure > 18 psig exceeds the limits of the recirculation filter capability.

D. Containment pressure greater than 18 psig could damage the recirculation filter fans when they are operated in the Filter Mode.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect but plausible. It is plausible that a "P" signal would prevent the containment recirculation fans from starting. The "P" signal does affect the containment structure cooling fans, as those fans trip on low CC flow due to a "P" signal. The student could have a

conceptual misunderstanding between the containment recirculation fan and containment structure cooling fan logics.

B. **Correct.** In the event of a LOCA containment structure recirculation is automatically actuated by a "P" signal. The "P" signal also prohibits the opening of filter dampers 34A and 34B and closing of recirc dampers 34C and 34D. Containment structure recirculation is designed to prevent stratification of gases associated with the LOCA.

C. Incorrect but plausible. It is plausible that the recirculation filters would have a limit on its capability to filter a radioactive release, however this is not the correct reason.

D. Incorrect but plausible. It is plausible that the recirculation filter fans could be damaged by drawing high starting amps if containment were significantly pressurized, however this is not the correct reason.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-Z.3, Response to High Containment Radiation Level (Rev 19) 1-NHY-503204, CAH Recirc Filter Fan Logic Diagram (Rev 5)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8038I04RO			(As available)
Question Source:	Bank #	X	101986	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
TEB22288

Examination Outline Cross-reference:	Level	RO	SRO
Q27	Tier #	1	
	Group #	2	
	K/A #	**W/E09** Natural Circulation Operations **EK1**. Knowledge of the operational implications of the following concepts as they apply to the (Natural Circulation Operations) **EK1.1** Components, capacity, and function of emergency systems.	
	Importance Rating	3.0	

Proposed Question:

Plant conditions:

- Reactor trip due to loss of offsite power.
- ES-0.2, "Natural Circulation Cooldown" is being performed.
- Pressurizer level is 35%.
- Charging flow is 55 GPM.
- Letdown flow is 70 GPM.
- The crew is rapid borating to cold shutdown boron concentration.
- One boric acid pump is running.
- CS-V-426, "Emergency Boration to Charging Pump Suction" is open.
- CS-FI-183, "Emergency Boration Flow" indicates 70 GPM.

Which action, if any, is required to properly perform rapid boration?

A. No action is required.
B. Reduce letdown flow.
C. Start an additional boric acid pump.
D. Raise charging flow greater than rapid boration flow.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. The student may have a backwards system conceptual error and believe

that maintaining charging flow less than rapid boration ensures sufficient injection of boron into the RCS.

B. Incorrect but plausible. With letdown flow greater than charging flow the student may belive that letdown flow should be reduced to prevent a VCT divert and associated removal of the higher borated RCS water.

C. Incorrect but plausible. The student may incorrectly believe that there is a minimum required flow rate for boration flow. In that case starting a second boric acid pump would increase the boration flow rate.

D. **Correct.** Per ES-0.2, "Natural Circulation Cooldown", step 2, charging flow must be maintained greater than rapid boration flow.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-0.2, Natural Circulation Cooldown (Rev 34)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1225I11RO			(As available)
Question Source:	Bank #	X	101985	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**10**	
	55.43		

Comments:
TEB31529

Examination Outline Cross-reference:	Level	RO	SRO
Q28	Tier #	2	
	Group #	1	
	K/A #	**003** Reactor Coolant Pump System (RCPS) **K3** Knowledge of the effect that a loss or malfunction of the RCPS will have on the following: **K3.03** Feedwater and emergency feedwater	
	Importance Rating		

Proposed Question:

Plant conditions:

- 25% power.
- The breaker for "A" RCP trips.
- The reactor does not trip.

Which of the following describes the effect on feed flow to the Steam Generators after transient stabilizes?

A. Feed flow goes down in all SGs.

B. Feed flow in SGs "B", "C" and "D" is unchanged; feed flow in SG A goes down.

C. Feed flow in SGs "B", "C" and "D" goes up; feed flow in SG A goes down.

D. Feed flow in SGs "B", "C" and "D" goes down; feed flow in SG A goes up.

Proposed Answer:	C

Explanation (Optional):

A. Incorrect. Increased steaming from the unaffected loops will cause feed flow to rise in those loops. Conversely, decreased steaming in the affected loop causes feed flow to go down. If the candidate believes power will go down, then it would be reasonable to assume feed flow will go down in all loops.

B. Incorrect. power in the unaffected loops goes up, but overall reactor power does not change. Failure to realize this makes the answer plausible.

C. **Correct**. Feed flow in the Steam Generators B, C and D goes up; feed flow

in Steam Generator A goes down D. Incorrect. Opposite of what actually happens.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1201.01, RCP Malfunction (Rev 18)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1181I05RO			(As available)
Question Source:	Bank #	X	101975	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2012 Diablo Canyon		
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	4		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q29	Tier #	2	
	Group #	1	
	K/A #	**004** Chemical and Volume Control System (CVCS) **K5** Knowledge of the operational implications of the following concepts as they apply to the CVCS: **K5.37** Effects of boron saturation on ion exchanger behavior	
	Importance Rating	2.6	

Proposed Question:

Plant conditions:

- A plant startup is in progress after 200 days of continuous operation.
- Reactor is at 2% power and stable.
- Condenser Steam dumps are in MANUAL.
- Tave is 557°F and stable
- Charging and letdown are matched.
- Rod control is in manual.

CS-TV-130, "Letdown Heat Exchanger Temperature Control Valve" cycles to FULL OPEN from 30% THROTTLED OPEN.

When the reactor stabilizes, reactor power will be _________ 2%, and Tave will be ________ 557°F.

A. greater than; equal to
B. greater than; greater than
C. less than; equal to
D. less than; less than

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. It is true that there would be an increase in reactor power (see description for answer B). It is plausible that the student would believe Tavg remains constant, as the question stem states that both rod control and steam dumps are in manual, however Tavg would increase (see description for answer B).

B. **Correct.** If CS-TV-130 cycles to FULL OPEN, the resulting letdown process flow temperature will be reduced. A reduction in letdown temperature will result in increased removal of boron in the letdown demineralizer. This will then result in a boron dilution of the reactor coolant. The dilution would introduce positive reactivity and a resulting increase in reactor power. Since both rod control and steam dumps are in manual, there would be an increase in Tavg as well.

C. Incorrect but plausible. It is plausible that the student would believe that there would be a decrease in reactor power if they misunderstood the concept of demineralizer boron absorption (see description for answer B). Additionally, it is plausible that the student would believe Tavg remains constant, as the question stem states that both rod control and steam dumps are in manual, however Tavg would increase (see description for answer B).

D. Incorrect but plausible. It is plausible that the student would believe that there would be a decrease in Tavg and a decrease in reactor power if they misunderstood the concept of demineralizer boron absorption (see description for answer B).

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-506197, CC-PAB Contn Enclosure Fan Rm & FSB Loop A Control Loop Diagram (Rev 11) 1-NHY-506270, CS-Therm Regen Letdown Reheat Hx and Letdown Hx Control Loop Diagram (Rev 14)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8024I08RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**5**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q30	Tier #	2	
	Group #	1	
	K/A #	**005** Residual Heat Removal System (RHRS) **K4** Knowledge of RHRS design feature(s) and/or interlock(s) which provide or the following: **K4.06** Function of RHR pump miniflow recirculation	
	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- MODE 5.
- "A" train RHR is operating in shut down cooling mode.
- "B" train RHR is aligned for shut down cooling and in Standby.
- "B" train of RHR is being placed in service in order to swap shut down cooling trains.

Which ONE of the following describes the operation of minimum flow valve RH-FCV-611 when the pump is started?

A. OPEN prior to pump start; CLOSES when pump flow reaches a setpoint of 750 GPM.

B. OPEN prior to pump start; CLOSES when pump flow reaches a setpoint of 1403 GPM.

C. CLOSED prior to pump start; OPENS when pump is running and flow is below 750 GPM; CLOSES when pump flow reaches a setpoint of 750 GPM.

D. CLOSED prior to pump start; OPENS when pump is running and flow is below 1403 GPM; CLOSES when pump flow reaches a setpoint of 1403 GPM.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible because 750 GPM is the setpoint for valve opening as flow is lowered, and also because the valve is open when the pump is shut off. This valve is ONLY interlocked with a flow switch, and not pump start circuitry.

B. **Correct.**

C. Incorrect. Would be correct if valve was interlocked with pump start circuitry, and plausible because plant systems have alignments where 'low flow' alarms are only active when a pump motor breaker is closed.

D. Incorrect. Would be correct if valve was interlocked with pump start circuitry, and plausible because plant systems have alignments where 'low flow' alarms are only active when a pump motor breaker is closed. In this case, the setpoint is correct.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	RHR Detailed Systems text Section 4.1.5, RHR Pump Recirculation Control (Rev 8)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8033I06RO			(As available)
Question Source:	Bank #	X	87912	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Wolf Creek		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q31	Tier #	2	
	Group #	1	
	K/A #	**006** Emergency Core Cooling System (ECCS) **K6** Knowledge of the effect of a loss or malfunction on the following will have on the ECCS: **K6.02** Core flood tanks (accumulators)	
	Importance Rating	3.4	

Proposed Question:

Plant conditions:

- 100% power
- "A" accumulator check valve was experiencing excessive back leakage.
- The crew stopped the leakage by closing and de-energizing SI-V-3, "A" accumulator isolation valve.
- T.S.3.5.1.1 has been entered for the inoperable accumulator and a plant shutdown has commenced.
- A cold leg LOCA occurs on "B" Reactor Coolant loop.
- RCS pressure drops to 40 psig.

Which of the following describes the response of the accumulators?

A. "B", "C" and "D" refill the core through the cold legs.

B. "C" and "D" refill the core and "B" back feeds to SGs and creates loop seal.

C. "C" and "D" refill the core and "B" accumulator dumps to containment floor.

D. "C" and "D" refill the core and "B" back feeds to the top of the core through SG to hot leg.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. It is true that the "C" and "D" accumulators would refill the core. It is plausible that the student may assume that the "B" accumulator injection line is configured downstream of the break (closer to the reactor vessel) and that accumulator water would flow to the core.

B. Incorrect but plausible. It is true that the "C" and "D" accumulators would refill the core. It is

plausible for the student to believe that the accumulator is associated with development of a "loop seal", as a loop seal dynamic is associated with a small sized cold leg break.

C. **Correct**. Given that the LOCA is a cold leg break, the "B" accumulator would dump to the containment building floor, as described in UFSAR, section 15.6.2.1, "Performance criteria for ECCS".

D. Incorrect but plausible. It is true that the "C" and "D" accumulators would refill the core. It is plausible that the student may assume that the "B" accumulator injection line is configured upstream of the break (closer to the intermediate leg) and that accumulator water would backflow to the core through the hot legs.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	UFSAR section 15.6.5.2.1, Accident Analyses, Decrease in Reactor Coolant inventory, Performance Criteria for ECCS (Rev 16)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1413I01RO				(As available)
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis		X		
10 CFR Part 55 Content:	55.41	**8**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q32	Tier #	2	
	Group #	1	
	K/A #	**006** Emergency Core Cooling System (ECCS) **2.1.27** Knowledge of system purpose and/or function.	
	Importance Rating	3.9	

Proposed Question:

Which of the following is an item described in the UFSAR as ECCS acceptance criteria?

A. The calculated hydrogen production from the cladding reaction with steam will not exceed the capabilities of the hydrogen removal system.

B. The zirc-water reaction postulated to occur at 1500°F will not become self-sustaining.

C. The calculated maximum fuel element cladding temperature shall not exceed 1100°F.

D. The total cladding oxidation shall not exceed 17% of the total cladding thickness.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. The containment hydrogen removal equipment does have limitations on its use that are based on containment hydrogen concentration, however the hydrogen removal equipment is not associated with any of the five ECCS acceptance criteria.

B. Incorrect but plausible. One of the ECCS Acceptance Criteria does pertain to zirc-water reactions, however that acceptance criteria reads "Peak Cladding Temperature <2200°F", which prevents a rapidly increasing zirc-water reaction.

C. Incorrect but plausible. One of the ECCS Acceptance Criteria does pertain to fuel cladding temperature, however that acceptance criteria states "Peak Cladding Temperature <2200°F". The 1100°F value stated in the distractor is a valid threshold number, however it is associated with the Core Cooling critical safety function (red path criteria) vice ECCS Acceptance Criteria.

D. **Correct.** Answer is correct per UFSAR, ECCS Acceptance Criteria.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	UFSAR section 15.6.5.1, Accident Analysis Decrease in Reactor Coolant Inventory (Rev 16)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L8034I03RO				(As available)
Question Source:	Bank #	X	TEB29624		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge	X			
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**8**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q33	Tier #	2	
	Group #	1	
	K/A #	**007** Pressurizer Relief Tank/Quench Tank System (PRTS) **K4** Knowledge of PRTS design feature(s) and/or interlock(s) which provide for the following: **K4.01** Quench tank cooling	
	Importance Rating	2.6	

Proposed Question:

Given the following condition:

- RC-TI-468, PRT Temperature indicator is 110°F and rising.
- RC-PI-469, PRT Pressure indicator is 8 psig and rising.

Which of the following describes how the PRT is cooled?

A. At 120°F the PRT automatically transfers to the RCDT through the RCDT heat exchanger using "A" PCCW to cool the heat exchanger.

B. At 120°F the PRT automatically recircs through the PRT heat exchanger, using "B" PCCW to cool the heat exchanger.

C. At 10 psig the PRT automatically transfers to the RCDT through the RCDT heat exchanger using "A" PCCW to cool the heat exchanger.

D. At 10 psig the PRT automatically recircs through the PRT heat exchanger, using "B" PCCW to cool the heat exchanger.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect but plausible. It is true that a) at 120°F RC-P-271, “PRT Pump” will start automatically, b) the PRT Pump has an available discharge flowpath to the RCDT, and c) the RCDT heat exchanger is cooled from train “A” PCCW. It is not true that the PRT contents will transfer to the RCDT system.

B. **Correct.** The PRT heat exchanger is cooled by train “B” PCCW. At 120°F RC-P-271, “PRT

Pump" will start automatically and recirc the PRT contents through the PRT heat exchanger via RC-V-309, "PRT Recirc/Transfer Isolation" valve.

C. Incorrect but plausible. It is true that a) the PRT Pump has an available discharge flowpath to the RCDT, and b) the RCDT heat exchanger is cooled from train "A" PCCW. It is not true that a) the PRT contents will transfer to the RCDT system, and b) the "PRT Pump" automatically starts based on PRT pressure.

D. Incorrect but plausible. It is true that a) the PRT will automatically recirc through the PRT heat exchanger, and b) the PRT heat exchanger is cooled with train "B" PCCW. It is not true that the "PRT Pump" automatically starts based on PRT pressure.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	VPRO D4474, PRT Temp High (Rev 02) PID-1-RC-B20846, Reactor Coolant System Pressurizer (Rev 14)

Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #			
	Modified Bank#	X	90098	(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55	55.41	7	

	55.43	

Comments:

2010 Comanche Peak

90098 prior to significant modification:

Given the following condition:
• Annunciator 1-ALB-05B, Window 2.3 - PRT HI TEMP has just alarmed.

Which ONE (1) of the following describes how the Pressurizer Relief Tank (PRT) is normally cooled per SOP-110A, Reactor Coolant Drain Tank System?

A: Recirculate the PRT through the Reactor Coolant Drain Tank Heat Exchanger, using Component Cooling Water to cool the Heat Exchanger.

B: Recirculate the PRT through the Reactor Coolant Drain Tank Heat Exchanger, using Reactor Makeup Water to cool the Heat Exchanger

C: Drain the PRT to the Reactor Coolant Drain Tank while making up to the PRT from the Demineralized Water Storage Tank

D: Drain the PRT to the Reactor Coolant Drain Tank while making up to the PRT from the Reactor Makeup Water Storage Tank

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Q34	Tier #	2	
	Group #	1	
	K/A #	**007** Pressurizer Relief Tank/Quench Tank System (PRTS) **A4** Ability to manually operate and/or monitor in the control room: **A4.10** Recognition of leaking PORV/code safety	
	Importance Rating	3.6	

Proposed Question:

The plant is at 81 % power.

A pressurizer code safety valve inadvertently lifts and remains partially open.

The following indications exist:

- Pressurizer pressure is 2205 psig and DECREASING.
- Temperature downstream of the safety valve indicates 276°F and INCREASING slowly
- PRT pressure is 47 psig and INCREASING

Which of the following is the reason for the temperature indication seen downstream of the safety valve?

A. The enthalpy of the saturated fluid in the pressurizer vapor space decreases rapidly when it becomes subcooled in the safety valve tailpipe.

B. The enthalpy of the saturated fluid in the pressurizer vapor space decreases as it loses energy due to the high-velocity head loss in the safety valve tailpipe.

C. The enthalpy of the saturated fluid in the pressurizer vapor space decreases as it passes through a safety valve, resulting in a temperature indication corresponding to the low-energy fluid in the tailpipe.

D. The enthalpy of the saturated fluid in the vapor space does not change as it passes through a safety valve, resulting in a temperature indication corresponding to the pressure in the PRT.

Proposed Answer:	D

Explanation (Optional):

A. Incorrect correct but plausible. All of the distractors are plausible as the question requires an understanding of the enthalpic process. Distractor "A" is incorrect because the fluid does not become subcooled in the tailpipe.

B. Incorrect but plausible. All of the distractors are plausible as the question requires an understanding of the enthalpic process. Distractor "B" is incorrect because enthalpy of the fluid has not changed in the safety valve tailpipe.

C. Incorrect but plausible. All of the distractors are plausible as the question requires an understanding of the enthalpic process. Distractor "C" is incorrect because enthalpy does not go down during a throttling process.

D. **Correct.** A leaking valve (throttling process) is a constant enthalpy process. If enthalpy does not change, you can follow the Mollier diagram across (left to right) to the new pressure, and follow the constant pressure line up to the saturation curve. The temperature indicated is the temperature of the vapor at the new pressure.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Steam Tables			
Proposed references to be provided to applicants during examination:				Steam Tables
Learning Objective:				(As available)
Question Source:	Bank #	X		
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Seabrook 2000		
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	7		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q35	Tier #	2	
	Group #	1	
	K/A #	**008** Component Cooling Water System (CCWS) **A1** Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CCWS controls including: **A1.04** Surge tank level	
	Importance Rating	3.1	

Proposed Question:

Plant conditions:

- 100% power.
- A 150 GPM tube leak initiates in the letdown heat exchanger.

Assuming no operator actions, which of the following events will occur?

A. RCP “A” and “D” overheat.

B. RCP “B” and “C” overheat.

C. Auto VCT divert to the PDT.

D. “A” PCCW head tank overflows.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is true that the “A” and “D” RCPs are cooled from Train “A” PCCW and it is true that a leak in the letdown heat exchanger would introduce higher temperature fluid into the PCCW loop, however that fluid would travel back to the PCCW heat exchanger prior to interfacing with the reactor coolant pumps.

B. Incorrect but plausible. It is true that a leak in the letdown heat exchanger would introduce higher temperature fluid into the PCCW loop, however that fluid would travel back to the PCCW heat exchanger prior to interfacing with the reactor coolant pumps, however the “B” and “C” RCPs are cooled from Train “B” PCCW. There is a common operator misconception with regard to recalling which RCP’s are cooled from which PCCW loop.

C. Incorrect but plausible. It is plausible that the VCT tank level would increase and eventually divert if the letdown heat exchanger leak resulted in PCCW leaking into the letdown system, however letdown system pressure is higher than PCCW system pressure.

D. **Correct.** Pressure on the letdown side of the letdown heat exchanger is higher than the PCCW side. A leak in the heat exchanger would cause letdown fluid to leak into the Train "A" PCCW loop, eventually resulting in overflow of the head tank.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	PCCW Detailed Systems text Figure 3.1, PCCW Loop "A" (Rev 9)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L803I04RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**14**	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q36	Tier #	2	
	Group #	1	
	K/A #	**010** Pressurizer Pressure Control System (PZR PCS) **K1** Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: **K1.08** PZR LCS	
	Importance Rating	3.2	

Proposed Question:

Given the following:

- 55% power.
- Power is being raised to 100% per OS1000.05, "Power Increase".
- PZR spray is forced with "C" and "D" Backup heaters.
- A TAVE instrument failure results in the instantaneous decrease of PZR level SETPOINT by 10%.

Which ONE of the following describes the resulting alarm and response of the PZR pressure control system?

A. PZR Level Deviation HIGH; ALL Backup Heaters energized.

B. PZR Level Deviation LOW; ALL Backup Heaters de-energized.

C. PZR Level Deviation HIGH; ALL Backup Heaters de-energized.

D. PZR Level Deviation LOW; ALL Backup Heaters energized.

Proposed Answer:	A

Explanation (Optional):

A. **Correct.** If the pressurizer setpoint instantaneously dropped by 10% then the level control system would detect that actual pressurizer level was 10% greater than the errant setpoint. The level deviation bistables would actuate a “PZR Level Deviation High” alarm and would energize all of the backup heaters.

B. Incorrect but plausible. The student may incorrectly believe that a “PZR Level Deviation Low”

signal is generated from the level setpoint decreasing. Based on that incorrect assumption it would then be plausible that the backup heaters would de-energize in anticipation of heater uncovery.

C. Incorrect but plausible. It is correct that the alarm received would be "PZR Level Deviation High". The student could incorrectly believe that the backup heaters are designed to de-energize to mitigate an increasing level transient.

D. Incorrect but plausible. The student may incorrectly believe that a "PZR Level Deviation Low" signal is generated from the level setpoint decreasing. The student could incorrectly believe that the backup heaters are designed to energize to mitigate a decreasing pressure transient.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509051, RC PZR Press and Lvl CTL W Functional Diagram (Rev 7) VPRO D4435, Pressurizer Level Deviation Low (Rev 03) VPRO D4436, PZR LVL Deviation High & BU Htrs On (Rev 03)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8027I06RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q37	Tier #	2	
	Group #	1	
	K/A #	**010** Pressurizer Pressure Control System (PZR PCS) **K6** Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: **K6.01** Pressure detection systems	
	Importance Rating	2.7	

Proposed Question:

The following plant conditions exist:

- 100% power.
- All control systems in automatic.
- PT-455/456 selected for Pressurizer Pressure control/backup.
- Pressurizer Pressure transmitter PT-455 fails HIGH.
- Assume no operator action.

What is the final plant condition?

A. Reactor tripped on pressurizer high pressure.

B. Both PORVs armed, no significant change in actual pressurizer pressure.

C. One PORV armed, heaters and sprays operating, slightly elevated pressurizer pressure.

D. Reactor tripped on low pressurizer pressure or OTΔT, SI actuated on Low pressurizer pressure.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. There is a high pressure reactor trip signal that comes from the pressurizer pressure instruments, however the trip signal coincidence is 2/4 channels >2385

psig.

B. Incorrect but plausible. It is true that pressurizer pressure instruments 455/456/457/458 supply arming and actuating signals to the PORV's, however the arming signals come from channels 457 and 458. A failure high of channel 455 would create an open demand signal for the "A" PORV, the PORV would not be armed, as channel 458 would still read actual pressurizer pressure. Also, it is plausible that there would be no significant change in actual pressurizer pressure if the student misunderstood the master pressure control circuit and believed that the pressurizer heaters would energize to counteract the opening spray valves.

C. Incorrect but plausible. It is plausible that a single PORV would arm, as the arming signal comes from a dedicated pressure channel, however the arming signals come specifically from channels 457 and 458 vice 455. Additionally, it is plausible for the student to believe that pressurizer pressure would be slightly elevated if they believed that the pressurizer heaters would energize and reach a balance with the spray valves (forced sprays).

D. **Correct.** PT-455 is the selected primary pressure transmitter feeding into the pressure control circuit. The master pressure controller output signal will wind up towards 100% output, which in turn will send a demand signal to both pressurizer spray valve controllers causing the spray valves to fully open. This would cause pressurizer pressure to continuously decrease, eventually causing a low pressure reactor trip and subsequent SI.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509051, RC PZR Press and Lvl CTL W Functional Diagram (Rev 7)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8027I06RO			(As available)
Question Source:	Bank #	X	101976	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
TEB20196

Examination Outline Cross-reference:	Level	RO	SRO
Q38	Tier #	2	
	Group #	1	
	K/A #	**012** Reactor Protection System **A3** Ability to monitor automatic operation of the RPS, including: **A3.07** Trip breakers	
	Importance Rating	4.0	

Proposed Question:

Plant conditions:

- 100% power.
- The Train "B" Reactor Trip Bypass breaker (BYB) is racked in and closed for testing.
- A 2/4 NIS Power Range High Flux (High Setpoint) reactor trip demand occurs.

How do the Train "A" Reactor Trip (RTA) and Train "B" Bypass (BYB) breakers respond?

A. RTA trips on UV only, BYB trips on UV only.
B. RTA trips on UV and shunt trips, BYB trips on UV only.
C. RTA trips on UV only, BYB trips on UV and shunt trips.
D. RTA trips on UV and shunt trips, BYB trips on UV and shunt trips.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect but plausible. There is a common operator misconception with regard to which combination of UV and shunt trips are associated with the RT and BY breakers. This statement applies to all of the distractors.

B. **Correct**. The reactor trip breaker will trip on both the UV and shunt trips. Bypass breakers only trip on UV.

C. Incorrect but plausible. There is a common operator misconception with regard to which

combination of UV and shunt trips are associated with the RT and BY breakers. This statement applies to all of the distractors.

D. Incorrect but plausible. There is a common operator misconception with regard to which combination of UV and shunt trips are associated with the RT and BY breakers. This statement applies to all of the distractors.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509042, Reactor Trip Signals W Functional Diagram (Rev 10)				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	L8056I10RO			(As available)	
Question Source:	Bank #	X	101991		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis			X	
10 CFR Part 55 Content:	55.41	7			
	55.43				
Comments: TEB32145					

Examination Outline Cross-reference:	Level	RO	SRO
Q39	Tier #	2	
	Group #	1	
	K/A #	**013** Engineered Safety Features Actuation System (ESFAS) **K4** Knowledge of ESFAS design feature(s) and/or interlock(s) which provide for the following: **K4.08** Redundancy	
	Importance Rating	3.1	

Proposed Question:

The following plant conditions exist:

- A plant heatup is in progress.
- RCS temperature is 380°F.
- RCS pressure is 1400 psig.
- All four (4) MSIV bypass valves and two (2) MSIVs (MS-V-86 & 90) are open.
- The operator is ready to open the remaining two (2) MSIVs.
- The MSIVs, bypass valves, and MSIV upstream drain valves all go closed.
- Safety injection is <u>not</u> actuated.

Which of the following conditions would have resulted in the above Main Steam valve alignment?

A. The operator placed the Train "A" MSI switch to ACTUATE.

B. The operator placed the Train "B" MSI switch to ACTUATE.

C. 2 of 3 pressure channels on at least one steam line indicated SG PRESS LO.

D. 2 of 3 pressure channels on at least one steam line indicated SG PRESS RATE HI.

Proposed Answer:	D

Explanation (Optional):

A. Incorrect but plausible. There is a common operator misconception with regard to what MSI signals close which steam line valves. In this case it is plausible that a Train "A" manual MSI signal would close all of the valves, particularly if the student confused the Train "A" manual and auto MSI signals, as a Train "A" automatic MSI signal does close all of the valves.

B. Incorrect but plausible. There is a common operator misconception with regard to what MSI signals close which steam line valves. The operator may believe that a Train "B" signal is specific to MS-V-86 and 90.

C. Incorrect but plausible. The question stem states that all MSIV's, MSIV bypass valves, and MSIV upstream drains go closed. The only signal that will close all of the valves is a Train "A" automatic MSI signal. A "SG PRESS LOW" automatic signal would accomplish that, however that automatic signal is blocked. The student could chose this distractor if they misunderstood the P-11 permissive and associated SI/MSI signal blocking/enabling functions.

D. **Correct.** The question stem states that RCS pressure is 1400 psig, which means that the P-11 signal is in its reset state and the low steam line pressure SI signal is blocked. In this configuration the automatic steam line isolation on high rate of steam line pressure drop is enabled, and the automatic steam line isolation low steam line pressure is blocked. The question stem states that all MSIV's, MSIV bypass valves, and MSIV upstream drains go closed. The only signal that will close all of the valves is a Train "A" automatic MSI signal. The automatic MSI signal would be generated from the enabled high rate of steam line pressure drop (SG PRESS RATE HI)

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509048, Safeguard Actuation Signals W Functional Diagram (Rev 17) 1-NHY-509047 sh 2, FW STM GEN Trip Signals W Functional Diagram (Rev 1)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8057I08RO, 10RO			(As available)
Question Source:	Bank #	X	101990	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
TEB20368

Examination Outline Cross-reference:	Level	RO	SRO
Q40	Tier #	2	
	Group #	1	
	K/A #	**013** Engineered Safety Features Actuation System (ESFAS) **K6** Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: **K6.01** Sensors and detectors	
	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- 100% power
- SI-PT-936, Containment Pressure (CH III) has been declared INOPERABLE.
- The following actions have been taken in accordance with tech. Specs:
  - The Hi-1 bistable has been tripped
  - The Hi-2 bistable has been tripped
  - The Hi-3 bistable has been bypassed

SI-PT-935, Containment Pressure (CH II) subsequently fails HIGH.

How will the plant respond to the PT-935 failure?

A. No automatic actuations will occur.

B. Only containment building spray will automatically actuate.

C. Only main steamline isolation and safety injection will automatically actuate.

D. Containment building spray, main steamline isolation, and safety injection will automatically actuate.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. The student may believe that the CBS, MSI, and SI signals all are generated from the Hi-3 signal. If that were the case then no automatic actuations would occur

(see explanation in answer "C" statement).

B. Incorrect but plausible. The student may incorrectly believe that the CBS actuation signal is generated from a Hi-2 signal, and the Hi-3 signal is associated with the "P" (Phase B Containment Isolation) signal. In this case, the student would believe that the CBS signal actuates.

   With regard to the Safety Injection signal, there are four conditions that will generate an SI signal, including Hi-1 containment pressure. If the student has not memorized Hi-1 is one of the four signals, then they would incorrectly believe that SI does not actuate.

   With regard to the Main Steam Isolation signal, there are four conditions that will generate a MSI signal, including Hi-2 containment pressure. If the student has not memorized Hi-2 is one of the four signals, then they would incorrectly believe that MSI does not actuate.

C. **Correct.** A CBS signal would be generated from the Hi-3 signal 2/4 channel coincidence. The stem of the question states that the Hi-3 bistable for PT-936 has been placed in bypass. The coincidence for the Hi-3 signal then becomes 2/3 of the remaining channels. With PT-935 failing high there is a 1/3 coincidence which does not actuate containment building spray.

   A Main Steamline Isolation signal is generated from the Hi-2 signal, 2/3 coincidence. The question stem states that SI-PT-936 Hi-2 bistable has been tripped. A high failure of any of the remaining channels actuates the Hi-2 signal. When SI-PT-935 fails high a Main Steamline Isolation is actuated.

   A Safety Injection signal is generated from the Hi-1 signal, 2/3 coincidence. The question stem states that SI-PT-936 Hi-1 bistable has been tripped. A high failure of any of the remaining channels actuates the Hi-1 signal. When SI-PT-935 fails high a Safety Injection signal is actuated.

D. Incorrect but plausible. The student may incorrectly believe that the CBS actuation signal is generated from a Hi-2 signal, and the Hi-3 signal is associated with the "P" (Phase B Containment Isolation) signal. In this case, the student would believe that the CBS signal actuates.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509048, Safeguards Actuation Signal W Functional Diagram (Rev 17)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L8035I13RO			(As available)
Question Source:	Bank #	X	88436	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Seabrook		

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<tr><td colspan="2">Comprehension or Analysis</td><td>X</td><td></td></tr>
<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="3">7</td></tr>
<tr><td>55.43</td><td colspan="3"></td></tr>
<tr><td colspan="5">Comments:</td></tr>
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Examination Outline Cross-reference:	Level	RO	SRO
Q41	Tier #	2	
	Group #	1	
	K/A #	**022** Containment Cooling System (CCS) **K2** Knowledge of power supplies to the following: **K2.01** Containment cooling fans	
	Importance Rating	3.0*	

Proposed Question:

Consider the following sequence of events:

- Containment Structure Cooling fans 1A,1B,1C,1D and 1F were in service.
- A complete loss off-site power occurred.
- Bus E-5 was restored by "A" EDG, and sequencing is complete.
- Bus E-6 failed to re-energize.
- RMO has not been reset.

What is the status of the Containment Structure Cooling fans?

A. No fans are running.
B. 1C, 1F fans are running.
C. 1A, 1B, 1D fans are running.
D. 1C, 1E and 1F fans are running.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect. Plausible as all fans are blocked from starting following LOP coincident with an SI.

B. **Correct**. CAH fans C, E and F are powered from US-53 from Bus 5.

C. Incorrect. CAH fans A, B and D are powered from US-63 form Bus 6.

D. Incorrect. Plausible as the control circuits would also start standby fan 1E if it were in Normal after Start. The standby fan is kept in PTL by procedure.					
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Containment HVAC Detailed System Text figure 4.3 (Rev 8)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L8038I04RO				(As available)
Question Source:	Bank #				
	Modified Bank#	X	TEB20525	(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55	55.41	7			

	55.43	

Comments:

TEB20525 prior to significant modification.
Consider the following sequence of events:

- Containment Structure Cooling fans 1A,1B,1C,1D,1F were in service.
- A Safety Injection actuated from 100%RTP.
- Containment pressure reached 16 PSIG and is now trending down.
- A complete loss off-site power occurred.
- Bus E-5 was restored by "A" EDG, and sequencing is complete.
- Bus E-6 failed to re-energize.
- No actuation signals have been reset.

What is the status of the Containment Structure Cooling fans?

A. 1A, 1B, 1D fans are running.

B. 1C, 1F fans are running.

C. No fans are running.

D. 1A, 1B, 1C, 1D, 1F fans are running.

Answer: C.

Examination Outline Cross-reference:	Level	RO	SRO
Q42	Tier #	2	
	Group #	1	
	K/A #	**026** Containment Spray System (CSS) **A4** Ability to manually operate and/or monitor in the control room: **A4.01** CSS controls	
	Importance Rating	4.5	

Proposed Question:

Plant conditions:

- A Safety Injection has occurred coincident with a Loss of Offsite power.
- The PSO is performing Attachment "A" of E-0, "Reactor Trip or Safety Injection.
- Containment pressure is 20 psig and slowly rising.
- Containment spray has not actuated for either train.

How does the PSO initiate "A" train containment spray?

A. Place both train "A" CBS/P/CVI switches in ACTUATE one at a time.

B. Place both train "A" CBS/P/CVI switches in ACTUATE simultaneously.

C. RESET RMO and then Place both train "A" CBS/P/CVI switches in ACTUATE one at a time.

D. RESET RMO and then Place both train "A" CBS/P/CVI switches in ACTUATE simultaneously.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. It is plausible that the two switches located together could be operated sequentially if the signal sealed in, however the circuitry/logic for manual CBS actuation requires that both switches be held to actuate simultaneously.

B. **Correct.** There are a total of 8 manual actuation switches on the main control board (4 train "A" switches and 4 train "B" switches). The two switches in the same train that are located

together must be actuated simultaneously to actuate that train of CBS. C. Incorrect but plausible. It is plausible that the two switches located together could be operated sequentially if the signal sealed in, however the circuitry/logic for manual CBS actuation requires that both switches be held to actuate simultaneously. Additionally, it is plausible that RMO would have to be reset, as is the case with many plant components, however for the CBS pump start circuitry the RMO contact is associated with a manual start of the pump. The auto start circuitry is in parallel with the normal start circuitry and is not affected by RMO. D. Incorrect but plausible. It is correct that both switches should be actuated simultaneously, however there is no need to reset RMO. The CBS pump start circuitry does have an RMO contact, however that contact is associated with a manual start of the pump. The auto start circuitry is in parallel with the normal start circuitry and is not affected by RMO.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509048, Safeguards Actuation Signal W Functional Diagram (Rev 17)

Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge	X		
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	7		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q43	Tier #	2	
	Group #	1	
	K/A #	**039** Main and Reheat Steam System (MRSS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.03** Indications and alarms for main steam and area radiation monitors (during SGTR)	
	Importance Rating		

Proposed Question:

Plant conditions:

- 100% power.
- SG levels are 50% and stable.
- RM-6548, Containment Alt gas radiation monitor is in HIGH alarm.
- RM-6481-1, "B" MS Line radiation monitor is in HIGH alarm.
- PZR level is 55% and lowering 5%/min.
- Letdown has been isolated.
- "A" Charging pump is operating.
- "B" charging pump is DTO.
- Charging flow is at 120 GPM.
- RCS pressure is 1950 psig and lowering.
- The crew is implementing the appropriate abnormal procedure(s).

Which of the following (1) specifies actions required and (2) which procedure is transitioned to when E-0, "Reactor Trip or Safety Injection" is exited?

A. (1) Actuate SI.
(2) E-3, "Steam Generator Tube Rupture".

B. (1) Actuate SI.
(2) E-1, "Loss of Reactor or Secondary Coolant".

C. (1) Trip the reactor then actuate SI.
(2) E-3, "Steam Generator Tube Rupture".

D. (1) Trip the reactor then actuate SI.
(2) E-1, "Loss of Reactor or Secondary Coolant".

Proposed Answer:	C	

Explanation (Optional):

A. Incorrect but plausible. It is true that the E-0 diagnostic step flowpath will direct a transition to E-3, "Steam Generator Tube Rupture". A transition to E-3 takes priority over a transition to E-1, "Loss of Reactor or Secondary Coolant". It is also true that a SI is warranted, however the procedural direction is to trip the reactor, verify that the reactor is tripped, and then actuate SI.

B. Incorrect but plausible. The question stem states that the "Containment Alt Gas Rad Monitor" is in High alarm. This condition is included in E-0, step 11, for checking the need for transition to E-1, "Loss of Reactor or Secondary Coolant", however, a transition to E-3 takes priority over a transition to E-1. Additionally, it is true that a SI is warranted, however the procedural direction is to trip the reactor, verify that the reactor is tripped, and then actuate SI.

C. **Correct**. Per OS1227.02,"Steam Generator Tube Leak, OAS item 1, "PZR Level Criteria", and step 2 RNO, the operators should attempt to maintain pressurizer level by reducing letdown, increasing charging flow, and starting a second charging pump as necessary. The question stem states that the "B" charging pump is unavailable. In this case pressurizer level cannot be maintained. The procedural direction is to then trip the reactor, verify that the reactor is tripped, and then actuate SI.

Additionally, it is true that the E-O diagnostic step flowpath will direct a transition to E-3, "Steam Generator Tube Rupture". A transition to E-3 takes priority over a transition to E-1, "Loss of Reactor or Secondary Coolant".

D. Incorrect but plausible. It is true that per OS1227.02,"Steam Generator Tube Leak, OAS item 1, "PZR Level Criteria", and step 2 RNO, the procedural direction is to trip the reactor, verify that the reactor is tripped, and then actuate SI. Additionally, the question stem states that the "Containment Alt Gas Rad Monitor" is in High alarm. This condition is included in E-0, step 11, for checking the need for transition to E-1, "Loss of Reactor or Secondary Coolant", however, a transition to E-3, "Steam Generator Tube Rupture" takes priority over a transition to E-1.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1227.02, Steam Generator Tube Leak (Rev 19) E-0, Reactor Trip or Safety Injection (Rev 50)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1190I03RO, L1202I10RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	

10 CFR Part 55 Content:	55.41	**10**
	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q44	Tier #	2	
	Group #	1	
	K/A #	**059** Main Feedwater (MFW) System **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.03** Overfeeding event	
	Importance Rating	2.7	

Proposed Question:

Given the following plant conditions:

- The plant is at 70% power.
- MS-PT-514, ""A" SG Pressure CH I" fails high.

Based on this failure, initially _________(1)_________overfed, and the BOP will take manual control of the ________(2)__________ to mitigate the consequences of the transient.

	_____(1)____	__________(2)____________
A.	ONLY the "A" SG will be	"A" FRV
B.	ALL SGs will be	"A" FRV and the MFP master speed control
C.	ONLY the "A" SG will be	"A" FRV and the MFP master speed control
D.	ALL SGs will be	ALL FRVs and the MFP master speed control

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. It is true that the “A” Steam Generator will initially be overfed, as feed water reg valve controller would sense that steam flow was higher than feed flow, however all of the steam generators will initially be overfed due to the feed pumps speeding up in response to a higher Δ pressure setpoint.

B. **Correct**. MS-PT-514 provides a density compensation signal for the SG “A” steam flow channel and also provides density compensation for the total steam flow value. The main

feedwater pump master speed controller utilizes total steam flow (plant power) to determine the programmed Δ pressure setpoint. The feed water reg valves utilize steam flow to create a steam flow-feed flow error signal.

If PT-514 fails high then all of the steam generators will initially be overfed due to the feedpumps speeding up in response to a higher Δ pressure setpoint. Additionally, the "A" Steam Generator would initially overfeed as the level controller senses steam flow as being higher than feed flow. he operator is required to take manual control of the MFP master controller and the "A" FRV.

C. Incorrect but plausible. It is true that the "A" Steam Generator will initially be overfed, as feedwater reg valve controller would sense that steam flow was higher than feed flow, however all of the steam generators will initially be overfed due to the feed pumps speeding up in response to a higher Δ pressure setpoint. It is true that the operator would take manual control of both the "A" FRV and the master speed control.

D. Incorrect but plausible. It is true that all of the steam generators would be overfed due to the feed pumps speeding up in response to a higher Δ pressure setpoint. It is true that the operator should take manual control of the MFP master speed control. It is plausible that the operator would take manual control of all of the FRVs to dampen all of the SG level transients, but this is not the case.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1235.04, SG Feed Flow-Steam Flow or Steam Pressure Instrument failure (Rev 14) 1-NHY-509013, SG 1&2 FD WTR & STM Flow, STM Pressure Process Control Block Diagram (Rev 4) 1-NHY-509033, SG Level Control Process Block Diagram (Rev 10) 1-NHY-509053, FW Control & Isolation W Functional Diagram (Rev 13) SGWLC Detailed system text fig 3.2 and 3.5 (Rev 4)

Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #	X	88261	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Watts Bar		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	4	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q45	Tier #	2	
	Group #	1	
	K/A #	**059** Main Feedwater (MFW) System **A3** Ability to monitor automatic operation of the MFW, including: **A3.03** Feedwater pump suction flow pressure	
	Importance Rating	2.5	

Proposed Question:

The following plant conditions exist:

- 85% power.
- Condenser back pressure is 5" Hg.
- FW-P-32A suction pressure is 210 psig. Speed is 5605 RPM.
- FW-P-32B suction pressure is 255 psig. Speed is 5790 RPM.

What is the expected Feedwater system response, if any?

A. None.

B. Only FW-P-32A trips.

C. Only FW-P-32B trips.

D. Both FW-P-32A and B trip.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. To answer this question correctly the student must correctly recall the low condenser vacuum, low pump suction pressure, and pump overspeed trip setpoints. All of the values given in the question stem are plausible.

B. **Correct.** The main feedwater pump low suction trip setpoint is 220 psig for each pump. FW-P-32A is below this setpoint. The condenser back pressure is 5" Hg, eq. to approx. 25" vacuum and is greater than the main feewater pump low vacuum trip setpoint is <18.5"hg. The main feed pump speeds (5605/5790 rpm) are less than the overspeed trip setpoints (5940/6060

rpm)

C. Incorrect but plausible. To answer this question correctly the student must correctly recall the low condenser vacuum, low pump suction pressure, and pump overspeed trip setpoints. All of the values given in the question stem are plausible.

D. Incorrect but plausible. To answer this question correctly the student must correctly recall the low condenser vacuum, low pump suction pressure, and pump overspeed trip setpoints. All of the values given in the question stem are plausible.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503592, FW Pump Turbine Pump Suct Press Trip Logic (rev13) FP700094 sh 3 of 45, A MFP Global Variable Report (rev 0) FP700100 sh 3 of 44, B MFP Global Variable Report (rev 0)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8062I08RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	TEB30807	(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55	55.41	**4**	

	55.43	

Comments:

TEB30807 Prior to significant modification:

The following plant conditions exist:

- 85% power.
- Condenser back pressure is 5” Hg.
- FW-P-32A suction pressure is 310 psig. Speed is 5605 RPM.
- FW-P-32B suction pressure is 255 psig. Speed is 5590 RPM.

What is the expected Feedwater system response, if any?

A. None.

B. Only FW-P-32A trips.

C. Only FW-P-32B trips.

D. Both FW-P-32A and B trip.

Answer: A

Examination Outline Cross-reference:	Level	RO	SRO
Q46	Tier #	2	
	Group #	1	
	K/A #	**061** Auxiliary / Emergency Feedwater (AFW) System **K6** Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: **K6.01** Controllers and positioners	
	Importance Rating	2.5	

Proposed Question:

Plant conditions:

- A reactor trip has occurred due to loss of offsite power.
- ES-0.1, "Reactor Trip response" is in progress.
- EFW flow to all SGs has been isolated to control SG level and RCS temperature.
- "B" EDG trips due to a fault on Bus 6.

What is the status of the ability to control SG levels?

A. All SGs can be controlled from the MCB.

B. All SGs can only be controlled by an operator in the EFW pump house.

C. "A" and "C" SGs can be controlled from the MCB. "B" and "D" SGs can only be controlled by an operator in the EFW pump house.

D. "B" and "D" SGs can be controlled from the MCB. "A" and "C" SGs can only be controlled by an operator in the EFW pump house.

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. All SG's do have EFW flow control valves that are powered from Bus 6, however, given the conditions in the question stem, the B and D SGs could not be controlled from the MCB because their Train "B" powered control valves ("apron valves") were closed prior to the "B" EDG tripping.

B. Incorrect but plausible. All SG's do have EFW flow control valves that are powered from Bus 6, however, given the conditions in the question stem, the "A" and "C" SGs could be controlled from the MCB because their Train "A" powered control valves ("apron valves") are still powered from the "A" EDG.

C. **Correct**. The question stem states that there was a loss of offsite power and that EFW flow to all four SGs was isolated. This means that the four EFW control valves on the main control board apron were taken to CLOSE and that the emergency busses were being powered by the emergency diesel generators. When the "B" EDG subsequently trips only the "A" and "C" EFW control valves are capable of being controlled from their control board switches. The "B" and "D" EFW control valves would have to be controlled locally utilizing the clutch level and manual valve wheel.

D. Incorrect but plausible. The question requires the student to know the train related power supplies to the two in-series EFW control valves for each SG. If the student had the power supplies reversed, then this would be the correct answer.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	UFSAR sect6.8, ESF Emergency Feedwater System pg 6 (Rev15)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L8045I01RO				(As available)
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q47	Tier #	2	
	Group #	1	
	K/A #	**062** A.C. Electrical Distribution **K3** Knowledge of the effect that a loss or malfunction of the ac distribution system will have on the following: **K3.01** Major system loads	
	Importance Rating	3.5	

Proposed Question:

Plant conditions:

- 100% power.
- The following alarm is received:

  D6329, BUS E5 UAT INC LN BKR TRIP & L/O

Which ONE of the following states the status of the "A" Diesel Generator?

The "A" Diesel Generator:

A. has started and sequenced loads that were previously running on Bus 5.

B. has started but does NOT load on Bus 5 until the Trip and L/O is reset.

C. does NOT start until the Trip and L/O is reset.

D. does NOT start until the Bus 5 UAT breaker is 'Green Flagged'.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. It is true that the EDG would start. It is plausible that the DG breaker could be closed as there are DG breaker related electrical protection devices that are bypassed (86DB) however they are bypassed on an SI signal.

B. **Correct**. The Bus E5 UAT Inc Line BKR Trip and L/O alarm indicates that the Bus 5 protection scheme is in place, locking out the "A" DG Breaker, however the condition does not prevent

the EDG from starting.

C. Incorrect but plausible. It is plausible that the EDG would not start as there is electrical protection (86DP) that would prevent the machine from starting. The second half of the answer is plausible as the lockout would have to be reset, however that would be for allowing breaker closure vice the DG start.

D. Incorrect but plausible. It is plausible that the EDG would not start as there is electrical protection (86DP) that would prevent the machine from starting.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	VPRO D6329, Bus E5 UAT INC LNBKR Trip and L/O (Rev 04)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8020I03RO			(As available)
Question Source:	Bank #	X	100755	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2012 St. Lucie		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q48	Tier #	2	
	Group #	1	
	K/A #	**062** A.C. Electrical Distribution **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the ac distribution system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.08** Consequences of exceeding voltage limitations	
	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- 100% power.
- It is 0200 on the mid shift.
- 345KV bus voltages are drifting up.
- 4160/480 bus voltages are also rising.
- D6001, "Vital UPS 1A INV SUPPLY VOLT HIGH" goes into alarm.

If voltages continue to rise what affect will this have on PP-1A and what actions are required, if any?

A. De-energizes. Reenergize from maintenance supply within 2 hours and restore power from Vital Inverter within 24 hours.

B. De-energizes. Reenergize from maintenance supply within 2 hours ONLY.

C. Remains energized from alternate AC source. No actions required.

D. Remains energized by Inverter 1A. No actions required.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is plausible that the alarm condition would automatically isolate the

inverter power supplies, de-energizing PP-1A, however the inverters DC supply will remain available and PP-1A will remain energized.

B. Incorrect but plausible. It is plausible that the alarm condition would automatically isolate the inverter power supplies, de-energizing PP-1A, however the inverters DC supply will remain available and PP-1A will remain energized.

C. Incorrect but plausible. It is plausible that PP-1A, being a safety related power source, would have a power supply auto swap feature, however the swap feature consists of a manual "break before make" breaker arrangement.

D. **Correct.** If the alarm setpoint is reached the AC supply breaker will trip, however inverters DC supply will remain available and PP-1A will remain energized.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	VAS VPRO D6001, Vital UPS 1A Inv Supply Volt High (Rev 02) T.S.3.8.3.1 OS1047.01, Vital Inverter Operation (Rev14) OS1426.05, DG 1B Monthly Operability Surveillance (Rev 30)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8018I03RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q49	Tier #	2	
	Group #	1	
	K/A #	**063** D.C. Electrical Distribution **K1** Knowledge of the physical connections and/or cause effect relationships between the DC electrical system and the following systems: **K1.03** Battery charger and battery	
	Importance Rating	2.9	

Proposed Question:

Given the following:

- 100% power.
- A bus fault causes a trip and lockout of the feeder to 480 volt MCC-512.
- The cause of the fault has not been determined.

Which ONE of the following describes the current power supply to PP-1A?

A. Energized from inverter 1A by DC Bus 11A, which is energized by Battery 1A.

B. Energized from inverter 1A by DC bus 11A, which is energized by Battery Charger 1A.

C. Energized from the automatic transfer to alternate AC power supply from MCC-531.

D. De-energized until manually transferred to the alternate AC power supply; from MCC-531.

Proposed Answer: A

Explanation (Optional):

A. **Correct**. MCC-512 feeds both Battery Charger 1A and Inverter 1A. If MCC-512 has a trip and lockout, DC Bus 11A will be fed from Battery 1A. Additionally, Inverter 1A will be fed from DC Bus 11A. Inverter 1A will continue to feed PP-1A.

B. Incorrect but plausible. It is true that PP-1A will be fed from Inverter 1A and that Inverter 1A will be fed from DC Bus 11A. It is plausible that Battery Charger 1A could have an alternate AC supply and continue to feed DC Bus 11A, however MCC-512 is it only power supply.

C. Incorrect but plausible. It is plausible that the power supply to PP-1A could include an automatic transfer switch, as is the case for PP-1E and PP-1F, however the power supplies for power panels 1A through 1D utilize a “break before make” manual transfer feature, as their associated inverters are asynchronous with the stations AC distribution system.

D. Incorrect but plausible. It is true that PP-1A utilizes a “break before make” manual transfer feature, as its associated inverter is asynchronous with the stations AC distribution system, however the power panel would remain energized from Inverter 1A.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	120VAC Detailed system Text Fig 5.1 (Rev 7)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8018I03RO			(As available)
Question Source:	Bank #	X	90553	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2011 Ginna		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q50	Tier #	2	
	Group #	1	
	K/A #	**064** Emergency Diesel Generators (ED/G) **K1** Knowledge of the physical connections and/or cause effect relationships between the ED/G system and the following systems: **K1.05** Starting air system	
	Importance Rating	3.4	

Proposed Question:

The following plant conditions exist:

- Bus E5 is de-energized.
- DG starting air receiver (TK-45A) discharge valve DG-V-61A is misaligned shut.
- Due to operator error, "A" EDG was tripped after successfully starting.
- Another start attempt is about to be made.

How will "A" DG respond?

A. Rolls but will not start.

B. Starts only if "A" DG starting air compressor C-2A is available.

C. Starts only if "A" DG backup air compressor C-18A is available.

D. Starts with the remaining air receiver as a source of starting air.

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is plausible that having only one receiver available would only provide the capacity for one engine start, however the remaining receiver contains enough air to start the diesel twice.

B. Incorrect but plausible. It is plausible that the remaining available receiver would have to re-pressurize via the air compressor prior to attempting another start, however the remaining receiver contains enough air to start the diesel twice.

C. Incorrect but plausible. It is plausible that the remaining available receiver would have to re-pressurize via backup air compressor 18A prior to attempting another start, however the remaining receiver contains enough air to start the diesel twice.

D. **Correct**. The capacity and configuration of the diesel air start system is such that if one air receiver should fail, the remaining receiver contains enough air to start the diesel twice.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1026.04, Operating DG 1A Starting Air System (Rev 12)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8019I05RO			(As available)
Question Source:	Bank #	X	101979	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:
TEB26752

Examination Outline Cross-reference:	Level	RO	SRO
Q51	Tier #	2	
	Group #	1	
	K/A #	**073** Process Radiation Monitoring (PRM) System. **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		

Proposed Question:

Given the following plant conditions:

- A release of the 'B' Waste Test Tank is in progress at 15 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:	B

Explanation (Optional):

A. 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-1458-1 if it were in service however the stem of the question states that the flow rate is 15 gpm. In this case 1-WL-FCV-1458-2 would be the in service valve.

B. **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 15 gpm so 1-WL-FCV-1458-2 would be the valve being utilized.

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

D. Incorrect but plausible. A High Radiation signal from RM-6509 will close the modulating Waste Test Tank Discharge Valve and will close 1-WL-FCV-1458-2. Additionally, the high radiation signal does not cause the Waste Test Tank pumps to trip.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-504062, WL-Liquid Effluent Discharge Logic Diagram (Rev 9)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8086I05			(As available)
Question Source:	Bank #	X	101359	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2010 Seabrook		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q52	Tier #	2	
	Group #	1	
	K/A #	**076** Service Water System (SWS) **A3** Ability to monitor automatic operation of the SWS, including: **A3.02** Emergency heat loads	
	Importance Rating	3.7	

Proposed Question:

Plant conditions:

- 100% power.
- An inadvertent Safety Injection occurs.
- The BOP is monitoring the MCB for response of Service Water (SW) system valves.

Which of the following is the proper SW valve response for these conditions?

	SW-V-16 and 18 DG HX outlet valves	SW-V-4 and 5 Isolation to Secondary heat loads
A.	Stroke open	Remain open
B.	Remain open	Stroke closed
C.	Remain open	Remain open
D.	Stroke open	Stroke closed

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is true that SW-V-16 and 18 will stroke open. It is plausible that SW-V-4 and 5 would remain open, for instance, if the student incorrectly believed that the main feed pumps needed continued cooling (MFP's trip on an SI signal).

B. Incorrect but plausible. It is true that SW-V-4 and 5 stroke closed on an SI signal. It is plausible that SW-V-16 and 18 would normally be open and remain open to ensure adequate cooling to the EDG's, however, SW-V-16 and 18 stroke open via the associated diesel engine's low

speed relay when engine speed reaches 125 rpm.

C. Incorrect but plausible. It is plausible that SW-V-4 and 5 would remain open, for instance, if the student incorrectly believed that the main feed pumps needed continued cooling (MFP's trip on an SI signal). Additionally, it is plausible that SW-V-4 and 5 would remain open, for instance, if the student incorrectly believed that the main feed pumps needed continued cooling (MFP's trip on an SI signal).

D. **Correct**. Both Emergency Diesel Generators will start on an SI signal. SW-V-16 and 18 will stroke open via the associated diesel engine's low speed relay when engine speed reaches 125 rpm. Additionally, SW-V-4 and 5 will stroke closed on an SI signal to allow full SW cooling capacity to primary system loads.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503496, DG- Diesel Start Logic Diagram (Rev 4) 1-NHY-503956, SW- SW to DG WTR Jacket HX Logic Diagram (Rev 6) 1-NHY-503977, SW- ISOL Valves for SW Flow to Turbine BLDG Logic Diagram (Rev 6)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8037I06RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	88757	(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Millstone 3		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55	55.41	7	

	55.43	

Comments:

88757 prior to significant modification:
With the plant initially at 100% power, a loss of offsite power occurs (the SAFETY INJECTION ACTUATION annunciator is NOT lit).

The RO is monitoring his boards to verify proper response of service water system valves.

Which service water system valves will the RO observe changing position, and to what position will they change?

A:

EDG cooling outlet valves	TPCCW HX supply valves	RPCCW HX supply valves
strokes open	remains open	strokes closed

B:

EDG cooling outlet valves	TPCCW HX supply valves	RPCCW HX supply valves
remains open	strokes closed	strokes closed

C:

EDG cooling outlet valves	TPCCW HX supply valves	RPCCW HX supply valves
remains open	remains open	remains open

D:

EDG cooling outlet valves	TPCCW HX supply valves	RPCCW HX supply valves
strokes open	strokes closed	remains open

Proposed Answer: D

Examination Outline Cross-reference:	Level	RO	SRO
Q53	Tier #	2	
	Group #	1	
	K/A #	**078** Instrument Air System (IAS) **K2** Knowledge of bus power supplies to the following: **K2.01** Instrument air compressor	
	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- 100% power.
- Service air compressors are aligned as follows:
  - SA-SKD-137A in LAG
  - SA-SKD-137B in LEAD
  - SA-SKD-137C in LAG
- A loss of offsite power occurs.
- "B" EDG fails to start.

What will be the status of instrument air system during the performance of ES-0.1, "Reactor Trip Response"?

A. SA-SKD-137B will be running as the LEAD compressor, SA-SKD-137A and 137C have been lost.

B. SA-SKD-137A air compressor will be running, SA-SKD-137B and 137C air compressor have been lost.

C. All air compressors are available, SA-SKD-137B will be running as the LEAD compressor.

D. All three electrically powered air compressors have been lost, but the Sullair compressor will maintain system pressure.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect but plausible. This question tests the student's knowledge of the service air compressor power supplies. This distractor is plausible if the student's assumptions are incorrect.

B. **Correct**. SA-SKD-137A is powered from US-52. SA-SKD-137B is powered from US-63. SA-SKD-137C is powered from US-21. Given the condition in the stem of the question, Bus E5 is energized via the "A" EDG. Bus E6 is not energized, as the "B" EDG failed to start. Busses 1-4 are de-energized as they have no backup power supply. Only SA-SKD-137A would remain running.

C. Incorrect but plausible. This question tests the student's knowledge of the service air compressor power supplies. This distractor is plausible if the student's assumptions are incorrect.

D. Incorrect but plausible. This question tests the student's knowledge of the service air compressor power supplies. This distractor is plausible if the student's assumptions are incorrect. Additionally, this distractor is plausible as the Sullair compressor is diesel driven, and would start if air pressure reduced to <100 psig.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1042.01, Operation of the Compressed Air System (Rev 29)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8023I13RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q54	Tier #	2	
	Group #	1	
	K/A #	**078** Instrument Air System (IAS) **K3** Knowledge of the effect that a loss or malfunction of the IAS will have on the following: **K3.01** Containment air system	
	Importance Rating	3.1*	

Proposed Question:

Both containment air compressors are out of service.

Containment air is being supplied from IA-V-530, Containment Instrument Air cross-connect.

Extended use of the Containment Instrument Air cross-connect supply at power will require more frequent ________________________.

A. operation of the Containment Air Purge system

B. containment entries to blow-down the IA receivers

C. operation of the Containment On-line Purge system

D. monitoring of the Containment Instrument Air moisture level

Proposed Answer: C

Explanation (Optional):

A. Incorrect but plausible. The CAP system is designed to purge the containment building, however it is used in the "Pre-Entry Purge" mode to support initial containment entry during outages, and in the "Refueling Purge" mode during outages.

B. Incorrect but plausible. It is plausible that the containment IA receivers would have to be blown down if the air being supplied via IA-V-530 had high moisture content, however the air supply is from the "outside containment" instrument air header "B", and has been conditioned via the IA system filters and dryers.

C. **Correct**. ON1042.02, "Operation of Containment Compressed Air System", Precaution 3.6 states "Use of the backup instrument air cross tie to containment (IA-V-530) should be limited to those occasions when the containment compressed air system is not capable of maintaining required air pressure. Minimizing the use of the cross tie will minimize the subsequent need for

COP system operation (to maintain containment pressure within limits)". Supplying containment IA via IA-V-530 will raise containment pressure because additional air is being added to the building without a means of removal.

D. Incorrect but plausible. It is plausible that the containment IA moisture level would have to be monitored if the air being supplied via IA-V-530 had high moisture content, however the air supply is from the "outside containment" instrument air header "B", and has been conditioned via the IA system filters and dryers.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1042.02, Operation of the Containment Compressed Air System (Rev 12)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8023I01RO			(As available)
Question Source:	Bank #	X	101980	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	7	
	55.43		

Comments:
TEB24407

Examination Outline Cross-reference:	Level	RO	SRO
Q55	Tier #	2	
	Group #	1	
	K/A #	**103** Containment System **A3** Ability to monitor automatic operation of the containment system, including: **A3.01** Containment isolation	
	Importance Rating	3.9	

Proposed Question:

An RCS leak resulted in the following conditions:

TIME	EVENT
0812	Manual Reactor Trip.
0826	Pressurizer Pressure 1850 psig and lowering.
0828	Manual Safety Injection.
0830	Pressurizer pressure 1800 psig and lowering
0907	Containment Pressure 4.3 psig and rising.
0941	Containment Pressure 18 psig and rising.
1003	RCS Pressure 220 psig and stable.

Assuming NO additional actions were taken, which ONE of the following choices describes the EARLIEST time a Containment Isolation signal was generated?

A. 0828

B. 0830

C. 0907

D. 0941

Proposed Answer: A

Explanation (Optional):

A. **Correct**. The Containment Phase "A" Isolation ("T" Signal) is actuated via a Safety Injection

signal (automatic or manual). The Containment Phase "B" Isolation ("P" Signal) is actuated via a Containment Building Spray signal (automatic or manual). At time 0828 a manual SI signal was actuated, which would in turn actuate the Containment Phase "A" Isolation ("T" Signal).

B. Incorrect but plausible. It is plausible that the student would incorrectly believe that only an automatic SI signal would actuate a Containment Phase "A" Isolation ("T" Signal). If this were the case, then the student could surmise that the Containment Phase "A" Isolation ("T" Signal) occurs when the Pressurizer Pressure Low SI setpoint (1800 psig) is reached at 0830.

C. Incorrect but plausible. It is plausible that the student would incorrectly believe that only an automatic SI signal would actuate a Containment Phase "A" Isolation ("T" Signal). If this were the case, then the student could surmise that the Containment Phase "A" Isolation ("T" Signal) occurs when the Containment Pressure Hi-1 setpoint (4.3 psig) is reached at 0907.

D. Incorrect but plausible. It is plausible that the student would incorrectly surmise that the Containment Phase "B" Isolation ("P" Signal) was first to occur a) if they misread the conditions in the question stem or b) they incorrectly believe that only an automatic SI signal would actuate a Containment Phase "A" Isolation ("T" Signal).

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509048, Safeguards Actuation Signal W Functional Diagram (Rev 17)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8057I10RO			(As available)
Question Source:	Bank #	X	88928	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2010 Ginna		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q56	Tier #	2	
	Group #	2	
	K/A #	**001** Control Rod Drive System **K1** Knowledge of the physical connections and/or cause effect relationships between the CRDS and the following systems: **KI.05** NIS and RPS	
	Importance Rating	4.5	

Proposed Question:

Plant conditions:

- 100% power.
- "B" train SSPS is in test.
- Rod control is in automatic.
- Rod height is 220 steps on Bank D.
- Power range channel N41 fails high.

How will the control rods respond?

A. Will not move.

B. Continuously insert unless stopped by the operator.

C. Insert until the power mismatch circuit output decays away.

D. Insert until the power mismatch circuit output decays away, then withdraw to match Average Tavg with Tref.

Proposed Answer:	C

Explanation (Optional):

A. Incorrect but plausible. It is plausible that the student could a) incorrectly surmise that the "C-2" control permissive blocks all automatic rod movement or b) have a misunderstanding of the automatic rod control signals.

B. Incorrect but plausible. It is plausible that the student could have a misunderstanding of the power mismatch circuit, incorrectly surmising that the error signal remained constant.

C. **Correct.** The control rod system creates an automatic rod movement signal based on a) temperature mismatch between RCS average Tavg and Tref (turbine impulse pressure), and b) power mismatch "rate of change" between auctioneered high NI power and turbine impulse pressure. The stem of the question states that NI power range channel N41 failed high. This condition would create a power mismatch signal causing control rods to insert. The "rate of change" signal output decays to zero over five 40 second time constants, which would cause inward rod movement to stop. There would also be a temperature error signal building in as RCS Tavg lowered during the rod insertion. Control permissive signal "C-2" blocks all manual and automatic rod withdrawal if 1 out of 4 NI power range instruments reads >103% power. With channel NI41 failed high, automatic rod withdrawal would be blocked.

D. Incorrect but plausible. Given the conditions in the question stem, as the power mismatch "rate of change" signal caused inward rod movement, the control rod system would create a corresponding temperature error signal warranting outward rod movement, however the "C-2" permissive signal would block automatic rod withdrawal. The student could either fail to recall the "C-2" permissive signal or have a conceptual error regarding the signal. In that case the student could incorrectly surmise that rods would eventually withdraw.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509049,Rod Control & Blocks W Functional Diagram (Rev 8)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L8031I15RO, L1182I09RO, 10RO				(As available)
Question Source:	Bank #	X	101982		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41	6			
	55.43				
Comments: TEB20749					

Examination Outline Cross-reference:	Level	RO	SRO
Q57	Tier #	2	
	Group #	2	
	K/A #	**014** Rod Position Indication System (RPIS) **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:

Just prior to a reactor startup with no failures, the following alarms are present:

- D7730, "One rod on bottom"
- D7749, "Two or more rods on bottom"

During the reactor startup when are these two alarms expected to reset?

A. When all shutdown rods are fully withdrawn and the bank overlap unit is manually reset.

B. When all shutdown rods are fully withdrawn and all control banks are withdrawn greater than 12 steps.

C. When all shutdown rods are fully withdrawn and control bank "A" rods are withdrawn greater than 12 steps.

D. When all control bank "D" rods are above the rod insertion limit and the bank overlap unit is manually reset.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. The bank overlap unit is reset prior to starting the reactor startup, but these alarms will only reset on actual rod heights greater than setpoint. Additionally, the alarms are based on DRPI indications, not group step counter demands.

B. Incorrect. The alarms will reset with control banks B, C, and D less than 12 steps provided they are above the minimum RIL

C. **Correct.** D7730 actuates when any rod in a withdrawn bank is on the bottom. DRPI "defines" a bank withdrawn as greater than 12 steps. D7749 actuates when at least 2 rods are less than 12 steps with the rod bottom bypass de-energized. This will reset when control bank A reaches the 12 step point D. Incorrect. Resetting the bank overlap unit prior to starting the reactor start up us required to correctly determine rod heights using GRPI, but the alarms are driven from DRPI indication. Additionally, the shutdown rods also would have to be withdrawn in order to reset these alarms.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Rod Control Detailed System Text section 5.2 (Rev 6)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8032I08RO			(As available)
Question Source:	Bank #	X	88453	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Seabrook		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**6**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q58	Tier #	2	
	Group #	2	
	K/A #	**015** Nuclear Instrumentation System **A1** Ability to predict and/or monitor changes in parameters to prevent exceeding design limits) associated with operating the NIS controls including: **A1.07** Changes in boron concentration	
	Importance Rating	3.3*	

Proposed Question:

Plant conditions:

- MODE 5 following refueling.
- SR N31/32 stable at 125/150 CPS respectively.
- A 500 gallon manual blended makeup to the VCT is in progress.
- The in service Boric Acid Tank pump manual discharge valve was inadvertently left shut following BAT swap over.
- Boric acid flow deviation alarm fails to activate.

What is the first alarm that will alert the operators and what actions will be directed to stop the inadvertent dilution?

A. Boron dilution monitor alarm. Stop the RMW pump.

B. Boron dilution monitor alarm. Stop the charging pump.

C. SR HI FLUX AT SHUTDOWN alarm. Stop the RMW pump.

D. SR HI FLUX AT SHUTDOWN alarm. Stop the charging pump.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** The boron dilution monitor setpoint is programmed to be 1.5 times the current calculated steady count value. Given the conditions in the question stem, the alarm setpoint would be approximately 225 CPS. The SR Hi Flux at Shutdown alarm setpoint is 750 CPS. In

the event of an inadvertent positive reactivity addition during plant shutdown, subcritical multiplication will cause the source range count rate to increase and eventually exceed the boron dilution monitor setpoint at approx. 225 CPS.

Additionally, per VAS Alarm Response Procedure D5972, the operators are directed to verify no RCS boron dilution is in progress, including verifying off/stopping the RMW pumps.

B. Incorrect but plausible. It is true that the boron dilution monitor would be the first alarm to alert the operators of the dilution event. Stopping the charging pump would stop the RCS dilution, as the VCT is the pump suction source, however per VAS Alarm Response Procedure D5972, the operators are directed to verify no RCS boron dilution is in progress, including verifying off/stopping the RMW pumps. The alarm response procedure does not direct securing the charging pump.

C. Incorrect but plausible. It is true that the correct action is to stop the RMW pump. The distractor is plausible as the student could have a misconception regarding the Boron Dilution Monitor alarm versus the Hi Flux at Shutdown alarm.

D. Incorrect but plausible. Stopping the charging pump would stop the RCS dilution, as the VCT is the pump suction source, however per VAS Alarm Response Procedure D5972, the operators are directed to verify no RCS boron dilution is in progress, including verifying off/stopping the RMW pumps. The alarm response procedure does not direct securing the charging pump. Additionally, the distractor is plausible as the student could have a misconception regarding the Boron Dilution Monitor alarm versus the Hi Flux at Shutdown alarm.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	VPRO D5972, Boron Dilution Monitor Trn B Alarm (Rev 06) NI Detailed System Text section 3.2.6 (Rev 6)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1410I06RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	1	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q59	Tier #	2	
	Group #	2	
	K/A #	**028** Hydrogen Recombiner and Purge Control System (HRPS) **K5** Knowledge of the operational implications of the following concepts as they apply to the HRPS: **K5.02** Flammable hydrogen concentration	
	Importance Rating	304	

Proposed Question:

A large break LOCA has occurred. The crew is evaluating placing the Hydrogen Recombiners in service.

As containment hydrogen concentration rises, ___________ is the maximum hydrogen concentration allowing the Hydrogen Recombiners to be placed in operation.

A. 4.0%

B. 5.0%

C. 6.0%

D. 7.0%

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Per OS1023.40, "Hydrogen Recombiner Operation", "Containment air hydrogen concentration **must** be kept below 4 percent by volume to prevent the possibility of an explosion in containment."

B. Incorrect but plausible. All distractors are numbers of reasonable value as compared to the correct answer, and are reasonable values of "lower explosive limits" for a multitude of various gasses.

C. Incorrect but plausible. All distractors are numbers of reasonable value as compared to the correct answer, and are reasonable values of "lower explosive limits" for a multitude of various gasses.

D. Incorrect but plausible. All distractors are numbers of reasonable value as compared to the correct answer, and are reasonable values of "lower explosive limits" for a multitude of various gasses.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1023.40 Hydrogen Recombiner Operation (Rev 08)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1203I02RO				(As available)
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge	X			
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q60	Tier #	2	
	Group #	2	
	K/A #	**029** Containment Purge System (CPS) **2.2.25** Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits	
	Importance Rating	3.2	

Proposed Question:

What is the basis for the T.S. limits on containment internal pressure?

A. Containment internal pressure does not exceed design pressure following a LOCA.

B. Minimize the effect of containment internal pressure on pressure instruments inside containment.

C. Normal containment leakage stays within the limits set by the Containment Leak Rate Testing Program.

D. Containment air lock seal leakage stays within limits set during surveillance testing of the air lock seals.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Per Tech. Spec. 3/4.6.1.4, Bases, The maximum peak pressure expected to be obtained from a LOCA event is 49.6 psig. The limit of 16.2 psia for initial positive containment pressure will limit the total pressure to 49.6 psig, which is less than the design pressure and is consistent with the safety analysis.

B. Incorrect but plausible. Pressure instruments reference legs are affected by atmospheric pressure, so it is plausible that the Tech. Spec. limit on containment pressure could be associated with maintaining pressure instrument accuracy.

C. Incorrect but plausible. Tech. Spec 3/4.6.1.2 covers containment leakage. The bases for that Tech. Spec. describes the need to maintain a limit on leakage. It is plausible that a limit on containment internal pressure would be pursuant to maintaining that “limit on leakage”.

D. Incorrect but plausible. Tech. Spec. 3/4.6.1.4 covers containment air locks. The bases for this specification explains that the limits on containment air lock leak rates are required to meets the restrictions on containment integrity. It is plausible that a limit on containment internal pressure would be pursuant to maintaining the "limits on containment air lock leak rates".

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S. Basis 3.6.1.4, Containment Internal Pressure, (Rev 118)				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	L8036I05RO			(As available)	
Question Source:	Bank #				
	Modified Bank#			(Note changes or attach Parent)	
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**9**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q61	Tier #	2	
	Group #	2	
	K/A #	**045** Main Turbine Generator (MT/G) System **K4** Knowledge of MT/G system design feature(s) and/or interlock(s) which provide for the following: **K4.02** Automatic shut of reheat stop valves as well as main control valves when tripping turbine	
	Importance Rating	2.5*	

Proposed Question:

Plant conditions:

- MODE 1.
- Power is being raised from 80% to 90%.
- MSRs have been placed in service.
- Due to a heater drain tank line up issue, MSRs are not draining correctly.
- "A" MSR level is 1 inch above the shell and rising.

At 3 inches above the shell which of the following depicts the correct valve alignment?

CV Control Valves

IV Intercept Valves

ISV Intermediate Stop Valves

MSR MS to MSR Isolation Valves

	CV	IV	ISV	MSR
A.	OPEN	OPEN	OPEN	OPEN
B.	CLOSED	OPEN	CLOSED	CLOSED
C.	CLOSED	CLOSED	CLOSED	OPEN
D.	CLOSED	CLOSED	CLOSED	CLOSED

Proposed Answer:	D	

Explanation (Optional):

A. Incorrect but plausible. The question relies on the student's memorization of the high MSR drain level turbine trip setpoint. The student may incorrectly believe that the trip setpoint has not been reached.

B. Incorrect but plausible. The student may incorrectly believe that the intermediate stop valves serves as the isolation for the cross-around steam side of the MSR's.

C. Incorrect but plausible. The MS to MSR Isolation valves are not associated with the turbine EHC system, thus student may incorrectly believe that the MSR valve does not trip closed on a turbine trip.

D. **Correct.** Given the conditions in the question stem, a turbine trip will occur. On a turbine trip, the electro-hydraulic control system dumps fluid pressure and closes the turbine control valves, stop valves, intercept valves and intermediate stop valves. Additionally, the MSR main steam supply valves will close.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503664, MS-Reheater Steam Supply Isolation Valves Logic Diagram (Rev 12) 1-NHY-503659, MD-High MSR Level Turbine Trip Logic Diagram (Rev 2)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8049I18RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**4**	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q62	Tier #	2	
	Group #	2	
	K/A #	**055** Condenser Air Removal System (CARS) **K1** Knowledge of the physical connections and/or cause-effect relationships between the CARS and the following systems: **K1.06** PRM system	
	Importance Rating	2.6	

Proposed Question:

Which of the following correctly describes immediate equipment response and/or required operator action associated with a high alarm on RM-6505, "Condenser Air Evacuation Radiation Monitor"?

A. All condenser Mechanical Vacuum Pumps will automatically trip.

B. All condenser Mechanical Vacuum Pumps will be shut off per OS1227.02, "Steam Generator Tube Leak".

C. AR-FV-5004, "Condenser Mechanical Vacuum Pump Discharge to ATM/PAB" automatically realigns to the PAB.

D. AR-FV-5004, "Condenser Mechanical Vacuum Pump Discharge to ATM/PAB" will be verified aligned to the PAB per OS1227.02, "Steam Generator Tube Leak".

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. In the event of a steam generator tube leak the mechanical vacuum pumps would draw radiation from the condenser, however there is no automatic pump trip signal generated from RM-6505, "Condenser Air Evacuation Radiation Monitor".

B. Incorrect but plausible. In the event of a steam generator tube leak the mechanical vacuum pumps would draw radiation from the condenser, however the guidance in OS1227.02, Attachment C is to verify that the vacuum pump discharge is aligned to the PAB vice stopping the pumps.

C. Incorrect but plausible. It is true that the desired vacuum pump discharge flowpath is to the PAB, however there is no automatic valve alignment signal generated from RM-6505, "Condenser Air Evacuation Radiation Monitor".

D. **Correct.** Per OS1227.02, Steam Generator Tube Leak", Attachment C, step 2 directs the operator to verify that the mechanical vacuum pump discharge is aligned to the PAB.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1227.02, Steam Generator Tube Leak (Rev 19)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1190I02RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		

10 CFR Part 55 Content:	55.41	7
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q63	Tier #	2	
	Group #	2	
	K/A #	**056** Condensate System **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	2.6	

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

Explanation (Optional):

A. 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.

B. **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.

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.

C. 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. 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.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1231.03, Turbine Runback/Setback (rev 21)				
Proposed references to be provided to applicants during examination:					NONE
Learning Objective:	L1183I09RO				(As available)
Question Source:	Bank #	X	101314		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2010 Seabrook			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q64	Tier #	2	
	Group #	2	
	K/A #	**071** Waste Gas Disposal System (WGDS) **K3** Knowledge of the effect that a loss or malfunction of the Waste Gas Disposal System will have on the following: **K3.05** ARM and PRM systems	
	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- 100% power.
- PAH FN-8A AND 8B are inadvertently secured.
- EOL dilution of 2000 gallons is in progress.
- Automatic divert of the VCT occurs due to large dilution.
- WG 3# header reaches 16 psig.

What are the consequences of this sequence of events?

A. Unmonitored release occurs.
B. WRGM radiation level increases.
C. WPB area radiation levels increase.
D. WPB airborne radiation levels increase.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Per VPRO B8219, “H2 Vent HDR Press Near Relief Setpoint”, “If the hydrogenated vent header reaches 15 psig, relief valves VG-V-82 and VG-V-83 will open, resulting in an unmonitored release out of the WPB roof.

B. Incorrect but plausible. It is plausible that the release path would be through the WRGM, however it is an unmonitored release.

C. Incorrect but plausible. The release path piping does pass through the WPB. Although pressurization of the 3# header was due to input of reactor coolant water into the PDT there would not be elevated WPB area radiation levels.

D. Incorrect but plausible. The release path piping does pass through the WPB. Although pressurization of the 3# header was due to input of reactor coolant water into the PDT there would not be elevated airborne radiation in the WPB as the release path is to atmosphere.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	WG Detailed System Text section 4.3.1 (Rev 4) 1-NHY-504014, VG-PAB Hydrogenated Vent Control Loop Diagram (Rev 4) 1-NHY-504015, VG-PAB Hydrogenated Vent HDR ISOL VLV Logic Diagram (Rev 1) 1-NHY-506873, VG-PAB Hydrogenated Vent HDR Isolation VLVs Control Loop Diagram (Rev 12)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8064I04RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**13**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q65	Tier #	2	
	Group #	2	
	K/A #	**079** Station Air System (SAS) **K4** Knowledge of SAS design feature(s) and/or interlock(s) which provide for the following: **K4.01** Cross-connect with IAS	
	Importance Rating	2.9	

Proposed Question:

Plant conditions:

- A piping rupture has occurred on a Service Air header.
- SA-C-137A and SA-C-137B have tripped.
- SA-C-137C is operating but cannot maintain system pressure.
- System pressure has dropped to 95 psig.

What automatic actions should already have occurred, if any?

A. None.

B. ONLY Sullair compressor started.

C. ONLY SA-V-92/93, “Service Air Isolation valves” closed.

D. Sullair compressor started AND SA-V-92/93, “Service Air Isolation valves” closed.

Proposed Answer:	B

Explanation (Optional):

A. Incorrect but plausible. This question requires memorization of setpoints for automatic features. Given the system pressure stated in the question stem, all of the distractors are plausible. With regard to distractor “A”, it is true that SA-V-92 and 93 would not have closed, however the Sullair compressor would have automatically started.

B. **Correct.** The Sullair compressor automatically starts when system pressure drops to 100 psig.

The compressor will then maintain system pressure at 115 psig. SA-V-92 and SA-V-93, "Service Air Isolation Valves" automatically close when system pressure crops to 90 psig. The valves have a re-open permissive setpoint of 95 psig increasing.

C. Incorrect but plausible. This question requires memorization of setpoints for automatic features. Given the system pressure stated in the question stem, all of the distractors are plausible. With regard to distractor "C", SA-V-92/93 would not have closed and the Sullair compressor would have started.

D. Incorrect but plausible. This question requires memorization of setpoints for automatic features. Given the system pressure stated in the question stem, all of the distractors are plausible. With regard to distractor "D", it is true that the Sullair compressor would have started, however SA-V-92/93 would not have closed.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1042.01, Operation of the Compressed Air System, figure 1 (Rev 29)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8023I16RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	7	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q66	Tier #	3	
	Group #		
	K/A #	**2.1.19** Ability to use plant computers to evaluate system or component status.	
	Importance Rating	3.9	

Proposed Question:

Plant conditions:

- An RCS leak is in progress.
- The leak rate has been determined to exceed T.S. limits.
- The crew is performing a down power of 25%/hour per OS1231.04, "Rapid Down Power".
- During the down power the BOP is monitoring leak rate by performing a RCS mass balance calculation.

The following data is displayed on the MPCS "GT RCSLEAK":

Time	A0332 PZR level	A0624 VCT level	A0316 TAVE
09:27:30	45.75%	37.00%	580.00°F
09:27:00	48.00%	35.00%	580.50°F

What is the RCS leak rate?

A. 30 GPM

B. 60 GPM

C. 120 GPM

D. 150 GPM

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Plausible if not multiplied by 2 since data display is on 30 second trend

B. **Correct.** PZR level decreases by 2.25% in 30 seconds. 60 gal/% in PZR. Multiply times 2 since

data is on 30 second trend = 270 gallon loss in 1 minute. VCT increases 2% in 30 seconds. 30 gal/% in VCT. Multiply times 2 = 120 gallon gained in 1 minute. Tave decreases 0.5°F in 30 seconds. 90 gal/°F. Multiply times 2 = 90 gallons. The RCS shrinks on temperature decrease so the expected shrink is added for the inventory balance. Total leakage in 1 minute = 270 - 120 - 90 = 60 gallons. C. Incorrect. Plausible if VCT level change is not multiplied by 2. D. Incorrect. Plausible if Tave change is interpreted incorrectly

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OX1401.02, RCS Steady State Leak Rate Calculation (Rev 08)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L3019I01RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	7		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q67	Tier #	3	
	Group #		
	K/A #	**2.1.29** Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	
	Importance Rating	4.1	

Proposed Question:

Which ONE (1) of the following methods is used to verify that a valve is LOCKED CLOSED per OS1090.05, "Component Configuration Control"?

A. Remove the lock, turn the valve in the CLOSE direction, and reinstall the locking device.

B. Remove the lock, turn the valve at least ¼ turn in the OPEN direction, close the valve, and reinstall the locking device.

C. Verify the locking device is installed, attempt to turn the valve in the OPEN direction, close the valve, and observe valve position indication.

D. Verify the locking device is installed, turn the valve in the CLOSE direction, and conduct a visual verification of the stem position

Proposed Answer: D

Explanation (Optional):

A. Incorrect but plausible. It is true that the valve should be checked in the closed direction, however the locking device should not be removed.

B. Incorrect but plausible. The locking device should not be removed, and manual valve position should be verified by turning the valve hand wheel in the closed direction.

C. Incorrect but plausible. It is true that the locking device should be checked installed and it is true that observation of valve position indication may be observed, provided that it is not the sole determinant of valve position, however valve position should be checked by turning the valve in the CLOSED position.

D. **Correct**. The operator verifies that the locking device is installed, and as stated in OS1090.05, Component Configuration Control, section 4.4.3, "To verify the position of a manual valve, always turn the valve hand wheel in the closed direction". Additionally, section 4.4.3 provides

guidance for observation of valve stem position, provided it is not the sole determinant of valve position.					
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1090.05, Component Configuration Control (Rev 58)				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:				(As available)	
Question Source:	Bank #	X	55590		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2010 Comanche Peak			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q68	Tier #	3	
	Group #		
	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. DRPI and demand positions agree within 6 steps.

Proposed Answer: C

Explanation (Optional):

A. 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. 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.

C. **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.

D. 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. OS1000.07 directs verifying that the DRPI and deman positions agree within 12 steps NOT 6 steps.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1000.07, Approach to Criticality (Rev 13)				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	L1162I02RO			(As available)	
Question Source:	Bank #	X	101284		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2010 Seabrook			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q69	Tier #	3	
	Group #		
	K/A #	**2.2.1** Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.	
	Importance Rating	4.5	

Proposed Question:

Plant conditions:

- Reactor Startup is in progress at EOL:
- Control Rods are in MANUAL.
- Reactor Power is 5%.
- Intermediate Range startup rate is 0 DPM.
- Steam Dump System is in the STEAM PRESSURE mode.
- PK-507, Steam Dump Pressure Controller is in AUTO with a setting of 8.40.

Which ONE (1) of the following would occur if PK-507, Steam Dump Pressure Controller potentiometer setting were to be changed to 8.70?

Tavg would __________ and Reactor power would __________.

A. decrease; increase
B. increase; decrease
C. increase; increase
D. decrease; decrease

Proposed Answer:	B

Explanation (Optional):

A. Incorrect. Plausible if thought that raising the potentiometer setpoint of the Steam Dump Pressure Controller would cause controlling steam pressure to lower. This would open the Steam Dump Valves and cause power increase.

B. **Correct**. Placing the Steam Dump Pressure Controller at a potentiometer setting of 8.70 would raise the controlling pressure in the Steam Generators (>1092 psig which corresponds to a potentiometer setting of 8.40) and hence, raise Tavg. With the core at end-of-life conditions and above the point of adding heat, negative reactivity is inserted and power will decrease.

C. Incorrect. Plausible because raising the potentiometer setpoint will cause Tavg to increase. If the core were at BOC conditions, positive reactivity could be inserted and power would increase.

D. Incorrect. Plausible if thought that raising the potentiometer setpoint of the Steam Dump Pressure Controller would cause controlling steam pressure to lower. If the core were at BOC conditions, negative reactivity could be inserted and power would decrease.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Steam Dump Detailed System Text pg 29 (Rev 5)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8047I06RO			(As available)
Question Source:	Bank #	X	97796	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Comanche Peak		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**1**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q70	Tier #	3	
	Group #		
	K/A #	**2.2.6** Knowledge of the process for making changes to procedures.	
	Importance Rating	3.0	

Proposed Question:

A situation arises during the performance of an operations procedure such that it cannot be performed as written. The correct action to be taken is to ________________.

A. initiate a procedure change while continuing with the work

B. stop the work and perform a procedure change before continuing

C. stop the work and have a supervisor make pen and ink changes before continuing

D. continue with the work and note any deviations for procedure enhancement following completion

Proposed Answer: B

Explanation (Optional):

A. Incorrect. Correct process but work must stop. You are allowed to continue with typo's, but not if procedure can't be performed as written.

B. **Correct.**

C. Incorrect. Pen and Ink changes are allowed for typo's but not for step changes.

D. Incorrect. Procedure enhancements can be made following completion. However not if procedure cannot be performed as written.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	AD-AA-100-1006, Procedure and Work Instruction Use and Adherence (Rev 5)

Proposed references to be provided to applicants during examination:					None
Learning Objective:	L5057I01RO				(As available)
Question Source:	Bank #	X	101390		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2009 Point Beach			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q71	Tier #	3	
	Group #		
	K/A #	**2.3.7** Ability to comply with radiation work permit requirements during normal or abnormal conditions.	
	Importance Rating	3.5	

Proposed Question:

The following conditions exist at a job site:

- The general area radiation levels are 40 mr/hr.
- Radiation level with shielding is 10 mr/hr.

Assumptions:

- Total time for one worker to install AND remove shielding is fifteen (15) minutes.
- Time to conduct the task with one worker is one (1) hour.
- Time to conduct the task with two workers is twenty (20) minutes
- Shielding is installed and removed by one (1) worker.

RWP Dosimetry set points:

- Rate 100 mr/hr
- Dose = 20 mr

In order to comply with radiation work permit requirements, which ONE of the following will result in the LOWEST whole body dose? Conduct the task with _____

A. One (1) worker with shielding.

B. Two (2) workers with shielding.

C. One (1) worker without shielding.

D. Two (2) workers without shielding.

Proposed Answer:	B

Explanation (Optional):

All distractors are plausible as the student could a) misinterpret/misapply the data provided in the stem. Additionally, the student could perform a mathematical error in any of the calculations.

A. Incorrect but plausible.
The dose for installing shielding is (.25 hr)(40 mr/hr) = 10mr
The dose for one worker to conduct the task with shielding is (1hr)(10mr/hr)=10mr
Total dose is 10mr + 10mr = 20mr

B. **Correct**.
The dose for installing shielding is (.25 hr)(40 mr/hr) = 10mr
The dose for two workers to conduct the task with shielding is (2)(.33 hr)(10 mr/hr) = 6.6 mr
Total dose is 10mr + 6.6mr = 16.6mr

C. Incorrect but plausible.
Total dose is (1hr)(40mr) = 40mr

D. Incorrect but plausible.
Total dose is (2)(.33hr)(40mr/hr) = 26.4mr

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	RP-AA-100-1002, Radiation Worker Instruction and Responsibilities (Rev 1)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1525I04RO, 05RO			(As available)
Question Source:	Bank #	X	100143	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2010 Beaver Valley		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	**12**	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q72	Tier #	3	
	Group #		
	K/A #	**2.3.15** Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.	
	Importance Rating	2.9	

Proposed Question:

Given the following plant conditions and sequence of events:

- 100% power.
- A small fuel element failure has just been confirmed.
- A Small Break LOCA inside containment occurs.
- SI actuates.

Fifteen (15) minutes has elapsed. Which of the following radiation monitors will accurately indicate radiation levels post trip?

1. RM-6526-2	Containment Recirc Noble Gas radiation monitor
2. RM-6520	Letdown radiation monitor
3. RM-6548	Containment Noble Gas Backup radiation monitor
4. RM-6576A/B	Post LOCA radiation monitors

A. 1 and 2 ONLY.

B. 2 and 3 ONLY.

C. 3 and 4 ONLY.

D. 1 and 4 ONLY.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. Initially, the containment atmosphere process radiation monitors will show increased

particulate and gaseous activity. The containment gas monitor is isolated upon a T signal and therefore will not accurately indicate post trip plant conditions. The trends prior to SI are useful for diagnosis. The letdown radiation monitor predominantly monitors for failed fuel and RCS crud burst. This monitor is also isolated upon a T signal and therefore although plausible is not available to accurately monitor post trip radiation levels unless letdown is un-isolated which is not the case 15 minutes post trip.

B. Incorrect. The containment backup gas will accurately indicate. The letdown radiation monitor will not accurately indicate because it is isolated on a T signal.

C. **Correct.** The Post LOCA monitor detectors are inside containment. They are safety related monitor with the RM-80s outside containment. The containment gas backup radiation monitor is located inside containment which monitors containment atmosphere and cannot be isolate on a T signal

D. Incorrect. Containment Gas monitor is incorrect for the reasons described above. The Post LOCA monitor is correct.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	DS Setpoint Summary "T" Signal Equipment Alignments (Rev June 2013)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8057I10RO			(As available)
Question Source:	Bank #	X	100142	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2010 Beaver Valley		
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41	**11**		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q73	Tier #	3	
	Group #		
	K/A #	**2.4.17** Knowledge of EOP terms and definitions.	
	Importance Rating	3.9	

Proposed Question:

Plant conditions:

- A Loss of Off-site power has occurred.
- No EFW pumps are available.
- RCS temperature is 560°F and increasing.
- RCS pressure is 1950 psig and increasing.
- The crew has entered FR-H.1, "Response to Loss of Secondary Heat Sink".

Which ONE of the following defines a "Dry Steam Generator" condition?

A. SG WR level is less than 14%.

B. SG WR level is less than 30%.

C. SG NR level is less than 6% with EFW flow to it less than 25 GPM.

D. SG NR level is less than 6% with EFW flow to it less than 100 GPM.

Proposed Answer: A

Explanation (Optional):

A. **Correct.** Per FR-H.1, Operator Action Summary Page, "Throughout this procedure, a "dry SG is defined as WR level less than 14% (30% for adverse containment)".

B. Incorrect but plausible. 30% is the defining value for "dry steam generator" for adverse containment. The question stem does not indicate adverse containment conditions.

C. Incorrect but plausible. 6% narrow range level is a defining criteria in FR-H.1, however it is the criteria for adequate SG level pursuant to exiting the procedure or securing bleed and feed (if in progress). Additionally, 25 gpm flow rate is not a criteria listed in FR-H.1

D. Incorrect but plausible. 6% narrow range level is a defining criteria in FR-H.1, however it is the criteria for adequate SG level pursuant to exiting the procedure or securing bleed and feed (if in progress). Additionally, 100 gpm is a defining criteria in FR-H.1, however it is the criteria for flow limit when feeding a dry SG with rising RCS temperature.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-H.1, Response to Loss of Secondary Heat Sink (Rev 35)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1211I02RO, 03RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	98727	(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2010 Diablo Canyon		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	10	
	55.43		

Comments:

98727 prior to significant modification:

What is meant by the term "Hot Dry" Steam Generator?

A: Primary Side Temperature >543°F and Wide Range level is less than 10% (18%)

B: Primary Side Temperature >550°F and Wide Range level is less than 10% (18%)

C: Primary Side Temperature >581°F and AFW flow has been isolated to the steam generator for an hour or longer

D: Primary Side Temperature >635°F and AFW flow has been isolated to the steam generator for an hour or longer

Proposed Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Q74	Tier #	3	
	Group #		
	K/A #	**2.4.23** Knowledge of the bases for prioritizing emergency procedure implementation during emergency operations.	
	Importance Rating	3.4	

Proposed Question:

Which ONE of the following describes the bases for why Functional Restoration Procedures (FRPs) are **NOT** implemented until specifically directed in ECA-0.0, "Loss of All AC Power" series?

A. All FRPs are written on the premise that at least one 4KV emergency bus is energized.

B. ECA-0.0 actions must be performed in sequence. Implementing FRPs interrupts the sequence and timing of steps.

C. ECA-0.0 includes all the key actions of RED path FRPs. Performing FRPs would be redundant and prolong the time until RCS depressurization was performed.

D. Certain diagnostic steps must be performed to minimize RCS leakage through the RCP seals. These steps are specific to ECA-0.0 and are not performed in any FRP.

Proposed Answer:	A

Explanation (Optional):

A. **Correct**. Westinghouse Background Document, ECA-0.0, page 84 states “The guideline has priority over all FRGs….. this priority is necessary since all FGRs are written on the premise that at least one ac emergency bus is energized”.

B. Incorrect but plausible. All of the distractors are plausible as they are written on the premise that the content/actions of ECA-0.0 take priority.

C. Incorrect but plausible. All of the distractors are plausible as they are written on the premise that the content/actions of ECA-0.0 take priority.

D. Incorrect but plausible. All of the distractors are plausible as they are written on the premise that the content/actions of ECA-0.0 take priority.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ECA-0.0 Background document. Pg 84 (Rev 2)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:					(As available)
Question Source:	Bank #	X	100147		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	Last NRC Exam	2010 Beaver Valley			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge			X	
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41	**10**			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q75	Tier #	3	
	Group #		
	K/A #	**2.4.34** Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.	
	Importance Rating	4.2	

Proposed Question:

OS1200.02B, "Remote Safe Shutdown Control - Train B" is being performed and "B" Train RSS operator is performing assigned attachments.

During the performance of the attachments the "B" Safety Injection pump is disabled by the operator.

(1) Where must the operator go to disable the "B" SI pump?

(2) What is the reason for disabling the SI pumps?

A. (1) CP-450B in the "B" EDG room.
   (2) Prevent SI pumps from starting, which could take RCS solid and make pressure control difficult.

B. (1) Bus-E6.
   (2) Prevent SI pumps from starting, which could take RCS solid and make pressure control difficult.

C. (1) CP-450B in the "B" EDG room.
   (2) Ensure SI pumps are available if needed for RCS inventory control.

D. (1) Bus-E6.
   (2) Ensure SI pumps are available if needed for RCS inventory control.

Proposed Answer: B

Explanation (Optional):

A. Incorrect but plausible. It is true that the reason for disabling the “B” SI Pump is for preventing a spurious pump start which could take the RCS solid. It is plausible that disabling the pump would be done at CP-450B, as that is a dedicated disabling panel with multiple components on

it.

B. **Correct.** Per OS1200.02B, "Remote Safe Shutdown Control – Train B", Attachment D, the SI pump is disabled by manually actuating the overcurrent relay at the pump breaker, which is located on Bus-E6. As described in EC145283, a spurious SI signal (or SI pump start) could result in a pressurizer overfill. During a plant cooldown this condition could result in the pressurizer going water solid and significantly impact RCS pressure control.

C. Incorrect but plausible. It is plausible that disabling the pump would be done at CP-450B, as that is a dedicated disabling panel with multiple components on it. It is also plausible that the SI pump is disabled to prevent a spurious start from damaging the pump and making it unavailable.

D. Incorrect but plausible. It is true that the SI pump is disabled at Bus-E6. It is plausible that the SI pump is disabled to prevent a spurious start from damaging the pump and making it unavailable.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1200.02B, Remote Safe Shutdown Control – Train B (Rev 18) EC145283, MSO Resolution: Procedures Only (Non-Hardware) Changes and Documentation Updates (Rev 2)

Proposed references to be provided to applicants during examination:				None
Learning Objective:				(As available)
Question Source:	Bank #	X	100896	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2012 Point Beach		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	**10**	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q76	Tier #		1
	Group #		1
	K/A #	**007** Reactor Trip **2.1.32** Ability to explain and apply system limits and precautions.	
	Importance Rating		4.0

Proposed Question:

A spurious reactor trip has occurred. The crew has processed E-0, "Reactor Trip or Safety Injection" and is performing ES-0.1, "Reactor Trip Response".

Which of the following describes the required operator actions regarding EFW throttling and the reason for those actions?

A. EFW pump recirculation valves must be opened prior to throttling EFW flow. The EFW pump recirculation valves were closed to meet the flow requirement for ANS Condition II events.

B. EFW pump recirculation valves must be closed prior to throttling EFW flow. The EFW pump recirculation valves were open to meet the flow requirement for ANS Condition II events.

C. EFW pump recirculation valves must be opened prior to throttling EFW flow. The EFW pump recirculation valves were closed to preclude potential stop check valve leakage rendering EFW pump steam voiding on auto initiation.

D. EFW pump recirculation valves must be closed prior to throttling EFW flow. The EFW pump recirculation valves were open to meet the minimum flow requirements for pump protection if feeding against SG pressure at the lowest set SG Safety valve setpoint.

Proposed Answer:	A

Explanation (Optional):

SRO Justification:

This question meets SRO level criteria 10CFR55.43(b)(2), Facility operating limitations in the tech specs and their bases. The question tests the examinee's "knowledge of TS bases that are required to analyze TS required actions and terminology". The question is a two part question, requiring the student to know a required action and the Tech. Spec. bases for the "as found" position of the components prior to the required action to reposition them.

A. **Correct.** As stated in ES-0.1, "Reactor Trip Response", OAS item 3, step 1 RNO, and step 6, the EFW pump recirculation valves are opened prior to throttling EFW flow. Additionally, bases for Tech. Spec. 3.7.1.2, "Auxiliary Feedwater System" lists multiple component and flow path configurations required for operability. This includes the statement "The EFW pump

recirculation valves must remain closed to meet the 650 gpm flow requirement for ANS Condition II events."

B. Incorrect but plausible. It is true that the requirements for EFW pump recirculation valve configuration is pursuant to meeting the flow requirements for ANS Condition II events, however the valves are initially in the closed position to meet this criteria.

C. Incorrect but plausible. It is true that the EFW throttle valves must be opened prior to throttling EFW flow, however the bases for the recirculation valve initial configuration is not associated with EFW pump steam voiding. Seabrook has had stop check valve leakage and EFW discharge Tee temperatures are checked on NSO rounds.

D. Incorrect but plausible. It is true that opening the EFW pump recirculation valves ensures flow requirements for pump protection, however the initial valve configuration is closed vice open. Additionally, the bases for the recirculation valve initial configuration is not associated with providing pump protection for feeding against SG pressure associated with the lowest SG safety valve setpoint.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-0.1, Reactor Trip Response (Rev 37) T.S. Basis 3.7.1.2, Turbine cycle AFW (Rev118)				
Proposed references to be provided to applicants during examination:					NONE
Learning Objective:	L1225I13RO				(As available)
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge	X			
	Comprehension or Analysis				
10 CFR Part 55 Content:	55.41				
	55.43	2			
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q77	Tier #		1
	Group #		1
	K/A #	**009** Small Break LOCA **EA2** Ability to determine or interpret the following as they apply to a small break LOCA: **EA2.02** Possible leak paths	
	Importance Rating		3.8

Proposed Question:

Plant conditions:

- MODE 3.
- SI actuated 15 minutes ago due to low PZR pressure.
- RCS pressure is 1200 psig and lowering.
- RWST is 450,000 gallons and lowering
- PZR level is off scale low.
- RCS Tave is 550°F and slowly lowering.
- RDMS indicates high radiation levels in the RHR vaults.

If the above conditions continue regardless of operator actions which one of the following depicts the correct procedure(s) and order the US will process?

A. OS1201.10, "Shutdown LOCA"

B. E-0, "Reactor Trip or Safety Injection"
   E-1, "Loss of Reactor or Secondary Coolant"
   ES-1.3, "Transfer to Cold Leg Recirculation"
   E-1, "Loss of Reactor or Secondary Coolant"

C. E-0, "Reactor Trip or Safety Injection"
   ECA-1.2, "LOCA Outside Containment"
   ECA-1.1, "Loss of Emergency Coolant Recirculation"

D. E-0, "Reactor Trip or Safety Injection"

E-1, "Loss of Reactor or Secondary Coolant"

ES-1.2, "Post LOCA Cooldown and Depressurization"

ES-1.3, "Transfer to Cold Leg Recirculation"

ES-1.2, "Post LOCA Cooldown and Depressurization"

Proposed Answer:	C	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Knowledge of diagnostic steps and decision points in the EOP's that involve transition to event specific sub-procedures or emergency contingency procedures. The question tests the student's ability to apply a plant condition (RCS pressure decreasing and RHR vault radiation) towards selection of the appropriate emergency contingency procedure (ECA-1.1).

A. Incorrect but plausible. It is plausible that the eventual mitigating procedure would be OS1201.10 Shutdown LOCA" as this procedure is applicable for lower Modes of operation and does contain steps for addressing loss of reactor coolant within the RHR system, however the procedure is only applicable in Mode 3 if the SI accumulators have been isolated.

B. Incorrect but plausible. If the student incorrectly believes that procedure E-1, "Loss of Reactor or Secondary Coolant" contains steps to mitigate a LOCA that interfaces with RHR, then it would seem plausible to remain in procedure E-1, transition to ES-1.3, "Transfer to Cold Leg Recirculation" when warranted, and then return to E-1, as would be the case with a RCS related LOCA.

C. **Correct.** Per E-0, "Reactor Trip or Safety Injection", once the diagnostic steps have been completed, and have indicated that there is no faulted SG, SG rupture, or RCS LOCA, then at step 20, the procedure directs a transition to "ECA-1.2, "LOCA Outside Containment" based on abnormal "Auxiliary Building Radiation". If the actions contained in ECA-1.2 are not successful, then that procedure directs a transition to ECA-1.1, "Loss of Emergency Coolant Recirculation".

D. Incorrect but plausible. If the student incorrectly believes that procedure E-1, "Loss of Reactor or Secondary Coolant" contains steps to mitigate RHR related LOCA's, then it would seem plausible to remain in procedure E-1 and transition to ES-1.2, "Post LOCA Cooldown and Depressurization".

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ECA-1.2, LOCA Outside Containment (Rev 25)				
Proposed references to be provided to applicants during examination:				NONE	
Learning Objective:	L1209I05RO			(As available)	
Question Source:	Bank #				
	Modified Bank#			(Note changes or attach Parent)	
	New	X			
Question History:	Last NRC Exam				
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)					
Question Cognitive Level:	Memory or Fundamental Knowledge				
	Comprehension or Analysis	X			
10 CFR Part 55 Content:	55.41				
	55.43	**5**			
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q78	Tier #		1
	Group #		1
	K/A #	**011** Large Break LOCA **2.4.6** Knowledge of EOP mitigation strategies.	
	Importance Rating		4.7

Proposed Question:

Plant conditions:

- A large break LOCA has occurred.
- E-1, "Loss of Reactor or Secondary Coolant" is in progress.
- Motor driven EFW pump has tripped.
- Turbine driven EFW pump operating at 270 GPM total flow with all EFW throttle valves fully open.
- All WR SG levels are 29% and DECREASING.
- A RED path is noted on the heat sink critical safety function status tree and you have transitioned to FR-H.1, "Response to Loss of Secondary Heat Sink".
- All SG pressures are 950 psig and STABLE.
- RCS pressure is 270 psig and DECREASING.
- RCS hot leg temps are 535°F and SLOWLY DECREASING.
- Containment pressure is 17 psig and INCREASING.

Which of the following actions will be directed next based on the above conditions and why?

A. Transition back to E-1. Total feed flow is less than 500 GPM.

B. Transition back to E-1. Core decay heat is being removed by the RCS break flow.

C. Remain in FR-H.1. Establish EFW flow greater than 500 GPM from SUFP to restore Secondary Heat Sink.

D. Remain in FR-H.1. Immediately perform steps 10-14, to initiate feed and bleed because 3 steam generators levels are less than 51% WR.

Proposed Answer:	B	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to apply a plant condition (RCS pressure less than SG pressures) towards selection of the appropriate procedure (E-1).

A. Incorrect but plausible. It is true that a transition back to procedure E-1 is warranted. It is also true that total EFW flow is less than 500 GPM which normally would be a reason to stay in FR-H.1, however RCS pressure is less than all of the non-faulted SG's, so FR-H.1 step 1 directs a transition back to procedure E-1.

B. **Correct.** Per FR-H.1, "Response to Loss of Secondary Heat Sink", step 1, if RCS pressure is less than all of the non-faulted SG's, then a transition back to procedure E-1 is directed. With RCS pressure less than non-faulted SG pressures then heat removal is via the RCS break.

C. Incorrect but plausible. It is plausible that the operators would take mitigative actions in FR-H.1 to establish >500 gpm feed flow, however RCS pressure is less than all of the non-faulted SG's, so FR-H.1 step 1 directs a transition back to procedure E-1.

D. Incorrect but plausible. Given the conditions in the question stem, FR-H.1 "feed and bleed" criteria are met. It is plausible that the operators would remain in FR-H.1 and initiate steps for feed and bleed, however RCS pressure is less than all of the non-faulted SG's, so FR-H.1 step 1 directs a transition back to procedure E-1.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-H.1, Response to Loss of Secondary Heat Sink (Rev 35)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1211I03RO			(As available)
Question Source:	Bank #	X	92042	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Seabrook		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**5**	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q79	Tier #		1
	Group #		1
	K/A #	**038** Steam Generator Tube Rupture (SGTR) **2.4.45** Ability to prioritize and interpret the significance of each annunciator or alarm.	
	Importance Rating		4.3

Proposed Question:

Plant conditions:

- A Steam Generator Tube Rupture has occurred in the "D" SG.
- The ruptured SG has been isolated.
- All RCPs are operating.
- Cooldown to target temperature of 490°F on the Steam Dumps is in progress.
- SI has been reset.
- RHR pumps have been placed in standby.

Subsequently the following alarms are received in sequence:

- MM-UA-53 B-3, SG 'B' LEVEL HI/LO
- RM-6482, 'B' Main Steam Line Radiation HIGH
- D4777, SG 'B' LEVEL HI-HI CHAN TRIP

What procedure action should be taken?

A. Actuate SI and return to E-3, "Steam Generator Tube Rupture" step 1.

B. Stabilize the plant and return to E-3, "Steam Generator Tube Rupture" step 1.

C. Transition to ECA-3.1, "SGTR with Loss of Reactor Coolant – Subcooled Recovery Required".

D. Transition to ECA-3.2, "SGTR with Loss of Reactor Coolant – Saturated Recovery Required".

Proposed Answer:	B

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. The question meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure with which to proceed. The student must assess the plant conditions given in the question stem, determine that there is an additional ruptured steam generator, and know to apply the Operator Action Summary Item "Multiple Tube Rupture Criteria". Correct application of the OAS item will direct the operators to the correct section of the procedure to identify and isolate the additional ruptured steam generator.

A. Incorrect but plausible. It is the correct action to return to step 1 of procedure E-3, however SI should not be actuated. OAS item 1, "ECCS Reinitiation Criteria" provides specific parameters used to determine the need to reinitiate ECCS. Should those criteria be met, the OAS item direction is to "manually align valves and start pumps as necessary" vice initiate an SI signal.

B. **Correct.** Per E-3, "Steam Generator Tube Rupture", Operator Action Summary Page item 4, "Multiple Tube Rupture Criteria", the correct action is to stabilize the plant and return to step 1 of the procedure, which guides the operators to the section of the procedure that identifies and isolates the ruptured generator.

C. Incorrect but plausible. E-3 does contain a multitude of decision point steps that direct a transition to ECA-3.1, however none of those decision points include the specific "multiple tube rupture criteria".

D. Incorrect but plausible. Given the fact that there would now be multiple tube ruptures, it is plausible that the student might think that the more aggressive mitigation strategy of ECA-3.2 might apply, however there is no guidance in E-3 that directs a transition to ECA-3.2

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-3, Steam Generator Tube Rupture OAS page. (Rev 42)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L3044I11SR			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55	55.41		

	55.43	**5**
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q80	Tier #		1
	Group #		1
	K/A #	**W/E05** Loss of Secondary Heat Sink **2.4.20** Knowledge of the operational implications of EOP warnings, cautions, and notes.	
	Importance Rating		4.3

Proposed Question:

Plant conditions:

- A Loss of Off-site power has occurred.
- No EFW flow is available.
- The crew has transitioned to FR-H.1, "Loss of Heat Sink" and is presently at step 3.
- The BOP operator reports that SG WR levels are:
  - SG "A": 25%
  - SG "B": 25%
  - SG "C": 35%
  - SG "D": 45%
- The following indications are noted:
  - RCS pressure is 2400 psig.
  - "A" PORV has opened in auto.
  - Hot leg temperatures are 645°F and increasing.
  - Cold leg temperatures are 640°F and increasing.

Which of the following summarize the actions taken next by the crew?

A. Transition to E-1 "Loss of Reactor or Secondary Coolant."

B. Attempt to establish EFW flow to at least one steam generator.

C. Attempt to establish SUFP flow to at least one steam generator.

D. Stop all Reactor Coolant Pumps and establish RCS bleed and feed.

Proposed Answer:	D

Explanation (Optional):
SRO Justification: This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to assess a plant condition(RCS pressure greater than 2385 psig due to loss of heat sink) and select a procedural strategy to mitigate the condition (initiate bleed and feed). A is incorrect but plausible. It is plausible that if the PORV is now open transitioning to E-1 is the correct procedure. When bleed and feed is stopped in FR-H.1, after regaining a heat sink, if a PORV cannot be closed transition to E-1 is directed. B is incorrect but plausible. Step 4 of FR-H.1 attempts to establish SUFP flow. This is incorrect as conditions require immediately commencing bleed and feed. C is incorrect but plausible. Step 3 of FR-H.1 attempts to establish EFW flow. This is incorrect as conditions require immediately commencing bleed and feed. D is **correct**. FR-H.1 has two conditions where bleed and feed are required to be performed any time after step 2. The Key Caution and Notes on the OAS page states "If wide range level in any 3 SGs is less than 30% [51% for adverse containment] OR PZR pressure is greater than or equal to 2385 PSIG due to loss of secondary heat sink, Steps 10 through 14 should be immediately initiated for bleed and feed (applicable after step 2)." Step 10 stops all RCPS and 11 through commence bleed and feed.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-H.1, Response to Loss of Secondary Heat Sink (Rev35)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1211I03RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41			
	55.43	**5**		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q81	Tier #		1
	Group #		1
	K/A #	**W/E11** Loss of Emergency Coolant Recirculation **EA2**. Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation) **EA2.1** Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	
	Importance Rating		4.2

Proposed Question:

Plant conditions:

- A Large Break LOCA has occurred.
- Emergency coolant recirculation capability has been lost.
- ECA-1.1, "Loss of Emergency Coolant Recirculation" is currently in progress.
- A RED path is identified on the CONTAINMENT status tree.
- The crew has transitioned to FR-Z.1, "Response to High Containment Pressure".

What procedure should be used to operate the containment spray pumps, and why?

A. FR-Z.1 because the RED path is higher priority than ECA-1.1.

B. FR-Z.1, because it provides for GREATER containment spray.

C. ECA-1.1, because an ECA should be completed PRIOR to transferring to an FR procedure.

D. ECA-1.1 because it provides for REDUCED containment spray.

Proposed Answer: D

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to apply a plant condition (loss of emergency recirculation capability) towards selection of the appropriate method of which to proceed with regard to operation of CBS pumps.

A is incorrect but plausible. Red path procedures have a higher priority than all other procedures except ECA-0.0 and ES-1.3. It is plausible actions required by FR-Z.1 would take priority over actions in ECA-1.1.

B is incorrect but plausible. Plausible as FR-Z.1 does require establishing maximum CBS flow in order to protect containment from high pressure. However reduced CBS flow is allowed if ECA-1.1 is in effect.

C is incorrect but plausible. Without the ability to continue to provide ECCS flow it could be decided that ECA-1.1 has a higher priority than FR-Z.1 and actions of ECA-1.1 are required to be completed prior to taking actions of FR-Z.1. This is incorrect as the Red path does have higher priority and it describes how to operate CBS pumps if ECA-1.1 is in effect.

D is **correct**. Due to the loss of emergency coolant recirculation ability, operation of CBS pumps is reduced to minimum required for containment conditions to preserve RWST water volume. FR-Z.1 has a caution prior to step 2 that states if ECA-1.1 is in effect the CBS pumps should be operated as directed per ECA-1.1 rather than FR-Z.1.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-Z.1, Response to High Containment Pressure (Rev23) ECA-1.1, Loss of Emergency Coolant Recirculation (Rev 36)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L12212I98RO			(As available)
Question Source:	Bank #	X	91476	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2008 Indian Point 2		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**5**	
Comments:			

Seabrook Station 2015 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q82	Tier #		1
	Group #		2
	K/A #	**037** Steam Generator (S/G) Tube Leak **AA2.** Ability to determine and interpret the following as they apply to the Steam Generator Tube Leak: **AA2.01** Unusual readings of the monitors; steps needed to verify readings	
	Importance Rating		3.4

Proposed Question:

Plant conditions:

- 100% power.
- Tave is 589°F and stable.
- The plant has been operating with a Steam Generator Tube Leak of 60 Gallon per Day (GPD) in the "B" SG for the last three months.
- OS1227.02, Steam Generator Tube Leak, is currently in use.
- The RDMS PRI>SEC leak monitor is temporarily unavailable.
- The following alarms are received within several minutes of each other:
  - ➢ RM 6505, Condenser Air Evac in Alert
  - ➢ RM-6511, "B" SGBD in Alert.

The leak rate has risen from 60 GPD to 90 GPD in the last 30 minutes.

Which of the following identifies the method to be used per OS1227.02, "Steam Generator Tube Leak" to determine the current leak rate?

A. Mass balance of VCT and PZR level.

B. Manual conversion of RM-6505 CPM to GPD.

C. Local low range radiation detection monitor on B MS line.

D. Flow balance of charging flow, letdown flow and seal return flow.

Proposed Answer:	B	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to apply a plant condition (steam generator tube leak rate) towards selection of the appropriate mitigation step. In this case the above actions are specified in Attachment "A" of OS1227.02.

A. Incorrect but plausible. When a SG tube leak is in the gallons per minute range, it is prudent for the operators to determine the leak rate utilizing a "mass balance" calculation and a "flow balance calculation. These methods are not prescribed in Attachment A and do not provide appropriate accuracy when the leak is in the gallons per day range.

B. **Correct.** Per OS1227.02, "Steam Generator Tube Leak", step 5, if tube leakage is less than 1 gpm the operator is directed to apply Attachment A, "Determine SG Leak Rate and Monitoring Requirements". The attachment provides two methods for determining the leak rate. One of those methods requires use of the PRI>SEC leak monitor. The question stem states that the monitor is unavailable. The alternate method is to perform a manual leak rate calculation by manually converting the RM-6505 reading to GPD, which utilizes a "correlation value" obtained from the "daily chemistry report".

C. Incorrect but plausible. OS1227.02 contains a note prior to step 3 that states "Local radiation monitoring using low range detectors may be required if RCS activity or steam generator tube leakage is very low". It is plausible that the local monitors could be used to calculate a "correlation value" however use of the local radiation detectors applies to step 3, which identifies the affected SG vice quantifies the leak.

D. Incorrect but plausible. When a SG tube leak is in the gallons per minute range, it is prudent for the operators to determine the leak rate utilizing a "mass balance" calculation and a "flow balance calculation. These methods are not prescribed in Attachment A and do not provide appropriate accuracy when the leak is in the gallons per day range.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1227.02, Steam Generator Tube Leak (Rev 19)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1190I04RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure

to provide the information will necessitate a detailed review of every question.)			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**5**	
Comments:			

Seabrook Station 2015 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q83	Tier #		1
	Group #		2
	K/A #	**074** Inadequate Core Cooling **2.4.18** Knowledge of the specific bases for EOPs.	
	Importance Rating		4.0

Proposed Question:

Plant conditions:

- A LOCA has occurred.
- NO ECCS equipment is available.
- Primary plant conditions:
  - CETCs 775°F and increasing.
  - RCS WR Pressure 1350 psig and stable.
  - All RCPS are OFF.
  - All SG NR levels 17% and slowly increasing.
- FR-C.1, "Response to Inadequate Core Cooling" has been entered.

Which of the following describes the method the RCS is depressurized and why?

A. Depressurize all SGs at maximum rate stopping at 125 psig. Rapidly lowers RCS pressure to allow accumulator injection.

B. Depressurize all SGs at maximum rate stopping at 125 psig. Minimize RCS inventory loss.

C. Depressurize RCS using ONE PORV. Rapidly lowers RCS pressure to allow accumulator injection.

D. Depressurize the RCS using ONE PORV. Minimize RCS inventory loss.

Proposed Answer: A

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant

conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to apply multiple plant conditions (ECCS equipment status, CETCs, RCP status, and SG levels) towards selection of the appropriate mitigation step.

A. **Correct**. Per FR-C.1, Response to Inadequate Core Cooling", step 11, all intact SGs are depressurized to 125 psig. Per the Westinghouse Background Document, FR-C.1, pg. 38, all SGs are depressurized to allow for SI accumulator injection.

B. Incorrect but plausible. It is true that all intact SGs are depressurized to 125 psig. It is plausible that the reason would be to reduce RCS inventory loss as lower pressure would decrease break flow.

C. Incorrect but plausible. It is true that the reason for the depressurization is to allow for SI accumulator injection. It is plausible that RCS depressurization would be accomplished via a PORV, as this would reduce RCS pressure, and is a method utilized in other EOP strategies, such as in E-3, "Steam Generator Tube Rupture" and FR-H.1, "Response to Loss of Secondary Heat Sink (bleed and feed).

D. Incorrect but plausible. It is plausible that RCS depressurization would be accomplished via a PORV, as this would reduce RCS pressure, and is a method utilized in other EOP strategies, such as in E-3, "Steam Generator Tube Rupture" and FR-H.1, "Response to Loss of Secondary Heat Sink (bleed and feed). Additionally, it is plausible that the reason would be to reduce RCS inventory loss as lower pressure would decrease break flow.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-C.1, Response to Inadequate Core Cooling (Rev 26)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1227I03RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41			
	55.43	5		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q84	Tier #		1
	Group #		2
	K/A #	**W/E03** LOCA Cooldown and Depressurization **2.4.4** Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.	
	Importance Rating		4.7

Proposed Question:

Following a LOCA, the crew entered E-1, "Loss of Reactor or Secondary Coolant".

The following plant conditions exist:

- Over the past 60 minutes Tcold has dropped 120°F and is currently 440°F and stable
- RCS pressure is 450 psig and stable
- S/G pressures are 580 psig and lowering slowly
- S/G Narrow Range levels are 40% - 45%.
- EFW flow is 300 GPM.
- Total injection flow is 1500 GPM.
- Pressurizer Level is 10% and rising slowly
- Containment pressure is 12 psig and lowering
- RWST level is 190,000 gallons and lowering

Based on the above conditions, which ONE of the following procedures will be transitioned to next?

A. ES-1.1, "SI Termination"

B. ES-1.2, "Post LOCA Cooldown and Depressurization"

C. ES-1.3, "Transfer To Cold Leg Recirculation"

D. FR-P.1, "Response to Imminent Pressurized Thermal Shock Condition"

Proposed Answer:	B
Explanation (Optional):	

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Knowledge of diagnostic steps and decision points in the EOP's that involve transition to event specific sub-procedures or emergency contingency procedures. The question tests the student's ability to apply plant condition (RCS subcooling <40°F) towards selection of the appropriate emergency sub-procedure (ES-1.2).

A. Incorrect but plausible. E-1, step 7 checks to see if ECCS flow should be reduced. If all of the criteria are met, then the step directs a transition to ES-1.1, "SI Termination". This transition is plausible if the student misinterprets or misapplies the conditions in the question stem. The subcooling and pressurizer level criteria for SI termination are not met.

B. **Correct**. Given the conditions in the question stem, the operators should navigate to E-1, "Loss of Reactor or Secondary Coolant", step 12, "Check if RCS Cooldown and Depressurization is Required". With RCS pressure greater than 300 psig, step 12 directs a transition to ES-1.2, "Post LOCA Cooldown and Depressurization".

C. Incorrect but plausible. Procedure E-1, OAS page item 5 does include criteria for transitioning to ES-1.3, "Transfer to Cold Leg Recirculation, however the criteria is not met.

D. Incorrect but plausible. The conditions in the question stem show that there has been a significant drop in RCS temperature. Per the FR-P critical safety function status tree, a transition to FR-P.1 is plausible, as the criteria for a 1 hour temperature decrease has been met, however the temperature vs. pressure (Figure 1) and RCS cold leg temperature <200°F Red Path criteria are not met.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-1, Loss of Reactor or Secondary Coolant (Rev 41)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1204I01RO			(As available)
Question Source:	Bank #	X	101042	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2011 Turkey Point		
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41			
	55.43	5		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q85	Tier #		1
	Group #		2
	K/A #	**W/E08** Pressurized Thermal Shock **EA2**. Ability to determine and interpret the following as they apply to the (Pressurized Thermal Shock **EA2.1** Facility conditions and selection of appropriate procedures during abnormal and emergency operations.	
	Importance Rating		4.2

Proposed Question:

Plant conditions:

- The plant was operating at 100% power.
- Main steamline break 15 minutes ago.
- Two SGs are depressurizing in an uncontrolled manner.
  - All RCS T_{hots} are 242°F and decreasing.
  - All RCS T_{colds} are 237°F and decreasing.
  - RCS pressure is 1200 psig and decreasing slowly.
  - E-2 Faulted Steam Generator Isolation step 3 is in progress.
  - ECCS flow is being supplied to the RCS.
  - Total EFW flow is 0 GPM.
  - NR level in the intact SGs is 92% and stable.
  - WR level in the affected SGs is 10% and decreasing.
  - Containment pressure is 13 psig and increasing slowly

Which procedure has highest priority, and why?

A. E-2 "Faulted Steam Generator Isolation". Two SGs are faulted. The faulted SGs must be isolated.

B. FR-H.1 "Response to Loss of Secondary Heat Sink". Loss of heat sink has occurred. The faulted SGs must be isolated.

C. FR-H.3 "Response to Steam Generator High Level". Both intact SGs are overfilled. Control of RCS pressure, temperature and SG level must be restored.

D. FR-P.1 “Response to Imminent Pressurized Thermal Shock Conditions”. Pressurized thermal shock is imminent. Control of RCS pressure, temperature and SG level must be restored.

Proposed Answer:	D	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria “Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed”. The question tests the student’s ability to apply multiple plant conditions (SG pressures, SG levels, RCS temperature, RCS pressure, EFW flow) towards selection of the appropriate procedure of which to proceed.

A. Incorrect but plausible. It is true that there are two faulted SGs and that procedure E-2 is the correct procedure to mitigate that condition, however the Red Path on the FR-P status tree warrants implementation of procedure FR-P.1, “Response to Imminent Pressurized Thermal Shock Conditions”. Per EOP rules of usage, implementation of the Red Path procedure takes priority.

B. Incorrect but plausible. It is true that the faulted SGs should be isolated (after implementation of FR-P.1). It is plausible that a transition to FR-H.1, “Response to Loss of Secondary Heat Sink” is warranted as SG NR and WR levels are the determining Red Path criteria, however with 2 NR levels greater than 6% the criteria is not met.

C. Incorrect but plausible. It is plausible that a transition to FR-H.3, “Response to Steam Generator High Level” is warranted as SG NR and WR levels are the determining Yellow Path FR-H.3 criteria, and that criteria is met. Implementation of FR-H.3 is not warranted as 1) the FR-P.1 procedure takes priority, and b) a Yellow Path FRP is discretionary.

D. **Correct.** Given the conditions in the question stem, there has been a cold leg temperature decrease of >100°F in the last 60 minutes and all RCS cold leg temperatures are less than 250°F. Per the CSF status tree for FR-P, the Red Path criteria are met and implementation of FR-P.1, “Response to Imminent Pressurized Thermal Shock Conditions” is warranted.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	F-0.3, Heat Sink (H) (Rev21) F-0.4, Integrity (P) (Rev 21)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L3046I11SR,12SR			(As available)
Question Source:	Bank #	X	101959	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure

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<tr><td rowspan="2">10 CFR Part 55 Content:</td><td>55.41</td><td colspan="3"></td></tr>
<tr><td>55.43</td><td colspan="3">5</td></tr>
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TEB28073</td></tr>
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Examination Outline Cross-reference:	Level	RO	SRO
Q86	Tier #		2
	Group #		1
	K/A #	**004** Chemical and Volume Control System (CVCS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the CVCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.09** High primary and/or secondary activity	
	Importance Rating		3.9

Proposed Question:

The following plant conditions exist:

- 100% power.
- 2 hours ago RM-6520, Letdown Radiation monitor went into HIGH alarm.
- VCT room and charging pump room area radiation monitors are in ALERT and radiation levels are increasing.
- Chemistry reports RCS DOSE EQUIVALENT I-131 activity is 0.6 μCuries per gram and slowly rising.
- Chemistry has confirmed the high activity with a second RCS sample.

The crew has entered the following procedures:

- OS1252.03, “Area High Radiation”.
- OS1252.01, “Process or Effluent High Radiation”.
- OS1202.05, “Reactor Coolant System High Activity”.

What action should be taken?

A. Verify letdown demins in service and adjust letdown to maximum rate.

B. Place the plant in Hot Standby with Tave less than 500°F within 6 hours.

C. Isolate/verify letdown has isolated automatically due to RM-6520 HIGH alarm.

D. Minimize personnel exposure by reducing charging and letdown to support RCP seal flow only.

Proposed Answer:	A	

Explanation (Optional):

SRO Justification:

This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to evaluate multiple plant conditions and multiple applicable AOP procedures in order to identify the correct procedurally prescribed mitigation strategy.

A. **Correct.** Per OS1202.05, "Reactor Coolant System High Activity", once chemistry has confirmed high activity with a second sample then the letdown demineralizers should be verified/placed in service and letdown flow should be maximized in order to maximize letdown decontamination factors.

B. Incorrect but plausible. If the RCS Dose Equivalent Iodine exceeded the T.S. 3.4.8 LCO, then step 4 of OS1202.05 would direct application of the T.S. action, which requires that the plant be placed in Hot Standby with Tavg <500 deg. F within 6 hours. The Dose Equivalent Iodine value given in the question stem is less than the Tech. Spec. LCO (1 μCI/gm), so no plant shutdown is necessary.

C. Incorrect but plausible. Per OS1252.01, "Process or Effluent High Radiation", if a process rad monitor is in High Alarm, then the procedure directs isolation of "release paths". It is plausible for the student to believe that the letdown system would be one of the systems that should be isolated. This is not the case as letdown is not a "release path" and the priority would be to maximize letdown per OS1202.05. Furthermore, the distractor states that letdown would automatically isolate due to RM-6520 being in high alarm, however there is no automatic isolation associated with that RM.

D. Incorrect but plausible. It is plausible that the operators would want to isolate letdown to minimize personal exposure, the priority would be to maximize letdown per OS1202.05.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1202.05, Reactor Coolant System High Activity (Rev 13)	
Proposed references to be provided to applicants during examination:		NONE
Learning Objective:	L1187I02RO	(As available)

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<tr><td>Question Source:</td><td>Bank #</td><td></td><td></td><td></td></tr>
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<tr><td>55.43</td><td colspan="3">5</td></tr>
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Examination Outline Cross-reference:	Level	RO	SRO
Q87	Tier #		2
	Group #		1
	K/A #	**039** Main and Reheat Steam System (MRSS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the MRSS; and (b) based on predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.04** Malfunctioning steam dump	
	Importance Rating		3.7

Proposed Question:

Given the following plant conditions:

- Reactor trip and Safety Injection from 100% power.
  - SG “A” ASDV is OPEN
  - SG “B” ASDV is CLOSED
  - SG “C” ASDV is CLOSED
  - SG “D” ASDV is CLOSED
- Steam Generator pressures:
  - SG “A”: 700 psig and 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 <u>rapidly</u> 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 to mitigate 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	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria “Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed”. The question tests the student’s ability to apply multiple plant conditions (SG pressure, level and Steam line radiation) towards selection of the appropriate sequence of procedures.

A. **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. 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. 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. 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): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection (Rev 50) E-2, Faulted Steam Generator Isolation (Rev 26) E-3, Steam Generator Tube Rupture (Rev 42)

Proposed references to be provided to applicants during examination:		NONE
Learning Objective:	L1202I09RO, 10RO	(As available)

Question Source:	Bank #	X		
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2010 Seabrook		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**5**	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q88	Tier #		2
	Group #		1
	K/A #	**061** Auxiliary / Emergency Feedwater (AFW) System **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.07** Air or MOV failure	
	Importance Rating		3.5

Proposed Question:

Given the following conditions:

- Tube rupture occurs on "A" SG.
- E-3, "Steam Generator Tube Rupture" is in progress.
- Flow from "A" Steam Generator has been isolated.

Which of the following failures has an impact on the Steam Generator isolation and what action should be taken to mitigate the situation?

A. 1) Airline severs to MS-V-393, "A" SG Steam Supply Valve to Turbine Driven EFW pump.
   2) Manually trip the Turbine Driven EFW Pump.

B. 1) "A" Train DC supply power is lost to "A" ASDV.
   2) Locally close "A" ASDV inlet isolation valve MS-V-5.

C. 1) Airline severs to MS-V-393, "A" SG Steam Supply Valve to Turbine Driven EFW pump.
   2) Locally close MS-V-393.

D. 1) "B" Train DC supply power is lost to "A" ASDV.
   2) Locally close "A" ASDV inlet isolation valve MS-V-5.

Proposed Answer:	C

Explanation (Optional):

SRO Justification:

This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria “Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed”. The question tests the student’s ability to address a plant condition Isolation of “A” SG in E-3 with a failed component (MS-V-393 failed open) apply an RNO to take actions to mitigate the failed component.

A. Incorrect but plausible. It is true that MS-V-393, “”A” SG Steam Supply Valve to Turbine Driven EFW Pump” fails open on a loss of air. Additionally, manually tripping the “Turbine Driven EFW Pump” would stop contaminated steam flow from exhausting to atmosphere, however procedure E-3, step 3 is designed to only isolate steam flow from the ruptured SG so that the pump remains running via the other steam supply.

B. Incorrect but plausible. It is true that procedure E-3, step 3 contains an RNO action for locally closing MS-V-5 in the event that the “A” ASDV cannot be closed from the main control board. It is plausible that the ASDV’s could fail open on loss of Train “A” DC power supply , however they do not.

C. Correct. MS-V-393, “”A” SG Steam Supply Valve to Turbine Driven EFW Pump” fails open on a loss of air. Procedure E-3, “Steam Generator Tube Rupture” step 3 contains actions for isolating flow from the ruptured SG. In the case of a failed open valve, the RNO for step 3 directs the operators to locally close the ruptured SG steam supply to the EFW pump.

D. Incorrect but plausible. It is true that procedure E-3, step 3 contains an RNO action for locally closing MS-V-5 in the event that the “A” ASDV cannot be closed from the main control board. It is plausible that the ASDV’s could fail open on loss of Train “B” DC power supply, however they do not.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-3, Steam Generator Tube Rupture (Rev 42) 1-NHY-506584, MS-PV-3001 Cont. Loop Dia. (Rev 6) 1-NHY-310841shE2T/8a, PV-3001 Schem. Dia. (Rev 8) 1-NHY-310841shE2U/15, PV-3001 Schem. Dia. (Rev 6) 1-NHY-506555, MS supply to TDEFW pump (Rev 25)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1205I03RO			(As available)
Question Source:	Bank #	X	87105	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Comanche Peak		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55	55.41		

	55.43	**5**
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q89	Tier #		2
	Group #		1
	K/A #	**064** Emergency Diesel Generators (ED/G) **2.2.22** Knowledge of limiting conditions for operations and safety limits.	
	Importance Rating		4.7

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- The motor driven Emergency Feedwater Pump has been removed from service for maintenance and is INOPERABLE.
- 30 minutes later the 'B' Charging Pump is declared INOPERABLE.
- An hour later the 'A' Emergency Diesel Generator is declared INOPERABLE.
- The Crew has entered Technical Specification 3.8.1.1, A.C. Sources.
- Per the applicable action statement the crew is demonstrating OPERABILITY of the remaining A.C. power sources.

Which of the following describes the correct interpretation of the ACTION requirements for Tech. Spec. 3.8.1.1, A.C. Sources?

A. The Tech. Spec. specifically requires that the motor driven Emergency Feedwater Pump be OPERABLE. Because the motor driven Emergency Feedwater Pump is INOPERABLE the associated ACTION is not met.

B. The Tech. Spec. requires that all required systems, trains, components and devices that rely on the remaining OPERABLE diesel generator are also OPERABLE. Because the motor driven Emergency Feedwater Pump and the 'B' Charging Pump are INOPERABLE the associated ACTION is not met.

C. The Tech. Spec. specifically requires that both the steam driven and motor driven Emergency Feedwater Pumps be OPERABLE. Because the motor driven Emergency Feedwater Pump is INOPERABLE the associated ACTION is not met.

D. The Tech. Spec. requires that all required systems, trains, components and devices that rely on the INOPERABLE diesel generator are also OPERABLE. There is no impact because the 'B' Charging Pump is associated with the OPERABLE diesel generator.

Proposed Answer:	B	

Explanation (Optional):

SRO Justification:

This question meets SRO level screening criteria 10CFR55.43(b)(2), Facility operating limitations in the technical specifications and their basis. This question requires the student to analyze plant conditions and apply T.S. 3.8.1.1 action (d) which has actions required to be met in 4 hours in addition to the 1 hour requirements.

A is incorrect but plausible. The Tech Spec requires the steam driven EFW pump to be operable. Per the Tech. Spec. basis, having an OPERABLE steam driven EFW pump ensures a diverse feed water supply to the steam generators. It is conceivable that the Tech. Spec. could specifically require the motor driven EFW in the case where the 'A' Emergency Diesel in INOPERABLE in order to meet the basis of ensuring feed water supply to the steam generators.

B is **correct**. Tech. Spec. 3.8.1.1, in addition to the 1 hour requirement for verifying offsite power sources, also requires:

1. All required systems, subsystems, trains, components, and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

2. When in MODES 1, 2 or 3, the steam driven emergency feed water pump is OPERABLE.

C is incorrect but plausible. The Tech Spec requires the steam driven EFW pump to be operable. Per the Tech. Spec. basis, having an OPERABLE steam driven EFW pump ensures a diverse feed water supply to the steam generators. It is conceivable that the candidate could think that the Tech. Spec. required operability of both EFW pumps to meet the basis of ensuring feed water supply to the steam generators.

D is incorrect but plausible. The Tech. Spec. requires that all required systems, trains, components and devices that rely on the OPERABLE diesel generator are also OPERABLE. The basis for this action is intended to provide insurance that a loss of offsite power condition does not result in a complete loss of safety function or critical features during a period when either diesel is inoperable. The candidate could misunderstand this basis and believe that the basis for the action is to prevent further vulnerability of the associated trains safety function or critical features.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.8.1.1, AC Sources Operating (Rev118)

Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L8011I25SR			(As available)
Question Source:	Bank #	X	101963	
	Modified Bank#			(Note changes or attach Parent)

	New			
Question History:	Last NRC Exam	2009 Seabrook		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	

10 CFR Part 55 Content:	55.41	
	55.43	**2**

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q90	Tier #		2
	Group #		1
	K/A #	**076** Service Water System (SWS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the SWS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.01** Loss of SWS	
	Importance Rating		3.7*

Proposed Question:

Plant conditions:

- 50% power.
- SW-P-41A is OOS for motor replacement.
- SW-P-41C trips on overcurrent.
- The crew is performing OS1216.01, "Degraded Ultimate Heat Sink".
- "A" Train Tower Actuation has been initiated.
- SW-V-20, "SW TRN "A" To Discharge Structure" cannot be closed.

What actions are required <u>at this time</u>?

A. Supply "A" train SW from SW-P-110B per attachment F of OS1216.01.

B. Trip the reactor. Enter E-0, "Reactor Trip or Safety Injection". After immediate actions stop "A" and "D" RCPs.

C. Place SW-P-110A in PTL within 10 minutes and commence plant shutdown per OS1231.04, "Rapid Down Power".

D. Close SW-V-75, "Turbine Building Discharge to CW" and commence emergency CT fill per OS1016.05, "Service Water Cooling Tower Operation".

Proposed Answer: C

Explanation (Optional):

SRO Justification:

This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility

conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to address a plant condition (SW-V-20 will not close) apply an RNO and select a specific section of a procedure in which to mitigate the condition.

A is incorrect but plausible. IF SW cooling were lost and could not be restored at lower modes when cooling to safety related loads is not required Attachment F provides direction to supply cooling from the opposite train cooling tower. This is not allowed while operating at power.

B is incorrect but plausible. Since cooling flow is lost to the "A" train PCCW system it is plausible the RCPs will overheat in a short time frame and a reactor trip is required. Direction in OS1216.01 is to monitor PCCW temperature and enter OS1212.01 PCCW System Malfunction. Reactor trip is only required if temperatures reach the trip value specified in OS1212.01.

C is **correct**. Per OS1216.01 if a cooling tower boundary isolation valve cannot be closed the affected cooling tower pump must be placed in PTL. This is a station time critical action to be performed within 10 minutes to minimize loss of cooling tower inventory and rendering both cooling tower trains inoperable. With no SW cooling to "A" train of PCCW, PCCW temperatures will be monitored and plant shutdown will be commenced while reducing heat loads on the SW system.

D is incorrect but plausible. It is plausible that realignment of the cooling tower discharge flow path could isolate the over board discharge of the cooling tower contents.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1216.01, Degraded Ultimate Heat Sink (Rev 23)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1193I02RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41			
	55.43	5		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q91	Tier #		2
	Group #		2
	K/A #	**002** Reactor Coolant System (RCS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.01** Loss of coolant inventory	
	Importance Rating		4.4

Proposed Question:

Given the following plant conditions:

- "B" EFW Pump is tagged out for motor replacement
- Plant tripped from an Inadvertent Safety Injection
- Offsite Power was subsequently lost
- "A" EDG failed to start
- The Turbine Driven EFW Pump tripped on overspeed
- FR-H.1, "Response to Loss of Secondary Heat Sink", is in progress
- RCS Bleed and Feed has been initiated

The following conditions exist:

- Secondary Heat Sink has been established by restoring the Turbine Driven EFW Pump
- RCS Bleed and Feed is being terminated
- The "A" PORV, can NOT be closed

Which ONE of the following identifies the correct procedure transition to be implemented?

A. E-1, "Loss of Reactor or Secondary Coolant".
B. Procedure and step in effect when FR-H.1 was entered.
C. ES-1.1, "SI Termination.
D. ES-1.2, "Post LOCA Cooldown and Depressurization.

Proposed Answer:	A	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to apply multiple plant conditions (bleed and feed terminated, PORV will not close) towards selection of the appropriate procedure. Additionally, the question tests the student's knowledge of a decision making point in an FRP that involves transition to a specific emergency operating procedure.

A. **Correct.** Per FR-H.1, step 27, if a PORV valve cannot be closed and its associated block valve cannot be closed then the procedure directs a transition to procedure E-1, "Loss of Reactor or Secondary Coolant".

B. Incorrect but plausible. There is a transition step in FR-H.1 that directs returning to the procedure and step in effect, however that is in the event that secondary heat sink has been restored and bleed and feed has not been established. The stem of the question does state that the secondary heat sink has been established, however bleed and feed is in progress.

C. Incorrect but plausible. Securing from feed and bleed does involve stopping ECCS pumps, however the steps for accomplishing these actions are contained in FR-H.1 vice transitioning to ES-1.1, "SI Termination". Additionally, the stem of the question states that a PORV is failed open, so transitioning to procedure E-1 takes priority.

D. Incorrect but plausible. A failed open PORV represents a LOCA condition, so it is plausible that, if the PORV cannot be closed, a cooldown and depressurization of the RCS would be used as a mitigation strategy, however FR-H.1 directs a transition to procedure E-1 vice ES-1.2, "Post LOCA Cooldown and Depressurization".

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-H.1, Response to Loss of Secondary Heat Sink (Rev 35)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1211I04RO			(As available)
Question Source:	Bank #	X	86928	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Harris		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	

10 CFR Part 55 Content:	55.41	
	55.43	**5**
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q92	Tier #		2
	Group #		2
	K/A #	**017** In-Core Temperature Monitor System (ITM) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the ITM system; and (b) based on those predictions, use procedures to correct, control or mitigate the consequences of those malfunctions or operations: **A2.02** Core damage	
	Importance Rating		4.1

Proposed Question:

Following a LOCA, the crew is performing the actions of E-1, "Loss of Reactor or Secondary Coolant", when the following conditions were noted:

- No RCPs are operating
- RCS pressure is 600 psig
- Core exit thermocouples (CETs) are reading 710°F
- PZR level is off scale low
- RVLIS Full Range indicates 0%

Based on these conditions, the unit supervisor should:

A. Transition to FR-C.1, "Response to Inadequate Core Cooling", because core uncovery is likely occurring.

B. Transition to FR-C.2, 'Response to Degraded Core Cooling", because core uncovery is likely occurring.

C. Transition to FR-C.1, "Response to Inadequate Core Cooling", because core damage is occurring.

D. Transition to FR-C.2, 'Response to Degraded Core Cooling", because core damage is occurring.

Proposed Answer: B

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Assessing plant conditions and then selecting a procedure or section of a procedure to mitigate, recover, or with which to proceed". The question tests the student's ability to apply multiple plant conditions (RCS temperature, RCS pressure, Rx vessel level) towards selection of the appropriate procedure of which to proceed. (FR-C.2)

A. Incorrect but plausible. This question requires knowledge of the specific FR-C critical safety function status tree criteria. If the student's knowledge of these criteria is incorrect, then they may surmise that procedure FR-C.1, "Response to Inadequate Core Cooling" should be implemented. Additionally, the basis stated in this distractor is correct, however it applies to FR-C.2 vice FR-C.1.

B. **Correct.** Per CSF flowchart F-0.2, "Core Cooling (C)", with no RCPs running, core exit thermocouples <725°F, and RVLIS full range level <40%, procedure FR-C.2, "Response to Degraded Core Cooling" should be implemented. Per the Westinghouse Background Document "FR-C.2 Response to Degraded Core Cooling", pg.2, "If the RCP's are not running, the degraded core cooling symptoms indicate the core is partially uncovered."

C. Incorrect but plausible. This question requires knowledge of the specific FR-C critical safety function status tree criteria. If the student's knowledge of these criteria is incorrect, then they may surmise that procedure FR-C.1, "Response to Inadequate Core Cooling" should be implemented. Additionally, this distractor includes the correct information relating to the basis for FR-C.1.

D. Incorrect but plausible. It is true that FR-C.2, "Response to Degraded Core Cooling" should be implemented, however the basis is that the core is partially uncovered. Symptoms related to core damage are associated with the basis for FR-C.1, "Response to Inadequate Core Cooling".

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	F-0.2, Core Cooling (C) (Rev 20)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1227I08RO			(As available)
Question Source:	Bank #	X	88517	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2008 South Texas		
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		

10 CFR Part 55 Content:	55.41	
	55.43	**5**
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q93	Tier #		2
	Group #		2
	K/A #	**068** Liquid Radwaste System (LRS) **2.2.44** Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	
	Importance Rating		4.4

Proposed Question:

Given the following conditions:

- The plant is at 100% power.
- A discharge of Waste Test Tank 'B' is in progress.
- The Liquid Radwaste Test Tank Discharge Radiation Monitor is then found to be failed low.

Which of the following describes the status of the Waste Test Tank discharge? (Reference material provided)

A. The discharge must be terminated and cannot recommence until the Liquid Radwaste Test Tank Discharge Radiation Monitor is returned to operable status.

B. The discharge must be terminated but can recommence provided samples are taken once per 24 hours if the specific activity is ≤ .01 microCurie/gram DOSE EQUIVILENT I-131.

C. The discharge must be terminated but can recommence provided the flow rate is estimated at least once per 4 hours during the actual release. Pump performance curves may be used to estimate flow.

D. The discharge must be terminated but can recommence provided two independent samples are analyzed and two technically qualified personnel independently verify the release rate calculations and discharge line valving.

Proposed Answer: D

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(2), Facility operating limitations in the technical specifications and their basis. The proposed question pertains to the Offsite Dose Calculation Manual (ODCM) which meets SRO screening guidance criteria.

A. Incorrect but plausible. There is an immediate action requirement to suspend a liquid rad release if the setpoint of the process radiation monitor is less conservative than ODCM Specification C.5.1. This is a known immediate action. The candidate could choose this answer if they do not properly apply ODCM Specification C.5.1. An improper setpoint is different than an inoperable monitor.

B. Incorrect but plausible. This distracter is associated with the action for an inoperable Steam Generator Blowdown Flash Tank Drain Radiation Monitor, which is also covered under specification table A.5.1-1.

C. Incorrect but plausible. This is the required action if the Liquid Radwaste Test Tank Discharge Radiation Monitor flow rate measuring device becomes inoperable. This action is also covered under specification table A.5.1-1.

D. **Correct.** ODCM Specification C.5.1, Radioactive Effluent Monitoring Instrumentation- Liquids, table A.5.1-1, action 29 states:

> "With the number of channels OPERABLE less than the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue provided that prior to initiating the release
> a) At least two independent samples are analyzed in accordance with Surveillance S.6.1.1, and
> b) At least two technically qualified members of the station staff independently verify the release rate calculations and discharge line valving.
> Otherwise, suspend release of radioactive effluents via this pathway."`

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ODCM Section A 5.1 (Rev 34)			
Proposed references to be provided to applicants during examination:				ODCM sect 5.1 pgs 1-10
Learning Objective:	L1512I05SR			(As available)
Question Source:	Bank #	X	101964	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis	X		
10 CFR Part 55 Content:	55.41			
	55.43	**2**		
Comments: TEB32293				

Examination Outline Cross-reference:	Level	RO	SRO
Q94	Tier #		3
	Group #		
	K/A #	**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:

Plant conditions:

- A Steam Line Break has occurred on Main Steam Line “A” and “D” inside containment.
- The crew has entered E-2, “Faulted Steam Generator Isolation”.
- The faulted SGs have been isolated and the crew is checking if ECCS flow should be reduced.
- The following conditions exist:
  - Containment pressure is 11 psig slowly increasing.
  - RCS pressure is 1770 psig and slowly increasing.
  - RCS temperature is 510°F and slowly lowering.
  - Total EFW flow is 400 GPM
  - WR level in all SGs are as follows:

    A=0% B=75% C=60% D=0%
  - NR level in all SGs are as follows

    A=0% B=20% C=5% D=0%
  - PZR level is 30% and increasing.

Which ONE of the following specifies the procedure transition required and why?

A. All criteria are met. Transition to ES-1.1, “SI Termination”.

B. A PZR level criterion is not met. Transition to E-1, “Loss of Reactor or Secondary Coolant”.

C. RCS subcooling criterion is not met. Transition to ES-1.2, “Post LOCA Cooldown and Depressurization”.

D. Secondary heat sink criteria are not met. Transition to FR-H.1, “Response to Loss of Secondary Heat Sink”.

Proposed Answer:	A	

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), assessment of facility conditions and selection of appropriate procedures. It meets the screening criteria "Knowledge of diagnostic steps and decision points in the EOP's that involve transition to event specific sub-procedures or emergency contingency procedures. The question tests the student's ability to apply plant conditions (RCS pressure and temperature, SG level, EFW flow and PZR level) towards selection of the appropriate sub procedure (ES-1.1).

A is **correct**. In E-2 step 7 subcooling, heat sink, RCS pressure, and PZR level are checked to determine if ECCS can be reduced. In the stem of the question containment is adverse (>4 psig). With adverse containment PZR level must be >28%, heat sink requires one SG NR level >15%, as well as subcooling >40°F and RCS pressure stable or increasing. Subcooling must be determined from conditions given in the stem using steam tables (109°F).

B is incorrect but plausible. With containment adverse the PZR level requirements are much greater than normally required. It is plausible 30% level does not meet this condition.

C is incorrect but plausible. The value for subcooling is not given. The student must use steam tables and given conditions to determine subcooling is greater than required. If not it is plausible that LOCA conditions require the strategy of ES-1.2 to mitigate

D is incorrect but plausible. The conditions for FR-H.1 are determined by EFW flow and SG water levels. EFW is required to be >500GPM or two SG WR levels >65% or one SG NR level >15%. EFW flow given is 400 GPM which is <500 GPM required, only one SG WR is >65% and one SG NR >15%.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-2, Faulted Steam Generator Isolation step 7. (Rev 26)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L1207I02RO			(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	

10 CFR Part 55 Content:	55.41	
	55.43	**5**
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q95	Tier #		3
	Group #		
	K/A #	**2.1.32** Ability to explain and apply system limits and precautions.	
	Importance Rating		4.0

Proposed Question:

Given:

The plant was performing an initial power ascension following a refueling outage.

After a 75 hour chemistry hold at 50% power, the plant commenced raising power at a steady rate of 2%/hour.

At 72% power two rods dropped and the crew performed a manual reactor trip.

Repairs to rod control required 15 days.

If the plant performed a startup and commenced power ascension from 20% to 30%, the power ascension limit in effect should be ___(1)___ to prevent fuel damage due to _____(2)_____. (Reference material provided.)

A. (1) 3%/hour
(2) pellet clad interaction

B. (1) 3%/hour
(2) radial power peaking

C. (1) 10%/hour
(2) pellet clad interaction

D. (1) 10%/hour
(2) radial power peaking

Proposed Answer: C

Explanation (Optional):

SRO Justification:

This question meets SRO level screening criteria 10CFR55.43(b)(6). Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming , and determination of various internal and external reactivity effects. Following core reloading there are administrative requirements required to condition fuel assemblies. These requirements are on power

ramp rates and hold points at various power levels.

A. Incorrect but plausible. The question stem states that the plant was at 50% power for 75 hours prior to the reactor trip. If the student misread the second block of Figure 1, item 5B, they could incorrectly surmise that the limiting ramp rate would be 3%/hr. Additionally, it is true that fuel precondition ramp rate limits are imposed pursuant to pellet clad interaction risk management.

B. Incorrect but plausible. The question stem states that the plant was at 50% power for 75 hours prior to the reactor trip. If the student misread the second block of Figure 1, item 5B, they could incorrectly surmise that the limiting ramp rate would be 3%/hr. Additionally, manipulation of control rods and core temperature profiles during power increase evolutions do effect core power distribution, however radial power peaking concerns are not associated with fuel preconditioning ramp rate guidelines.

C. Correct. Given the conditions in the question stem, OS1000.05, "Power Increase", Figure 1: Limitations and Setpoints, item 5b, "Ramp Rate Limitations" directs the operator to refer to Figure 1, item 6b. Item 6b prescribes a maximum rate of power increase of 10%/hr when power is between 20% and Pmax, which is 50% power. Additionally, per Westinghouse WCAP-17069-P, "Startup and Conditioning Basis for Westinghouse PWR Fuel", section 4.7, "PCI Risk Management During Reactor Operations" describes threshold power levels, allowable ramp rates etc., pursuant to pellet clad interaction risk management.

D. Incorrect but plausible. It is true that the limiting power ramp rate is 10%/hr. Additionally, manipulation of control rods and core temperature profiles during power increase evolutions do effect core power distribution, however radial power peaking concerns are not associated with fuel preconditioning ramp rate guidelines.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1000.05, Power Increase (Rev 27)

Proposed references to be provided to applicants during examination:	OS1000.05 fig 1 (Sh 2 and 3 of 3)

Learning Objective:	L1158I10RO			(As available)
Question Source:	Bank #	X	101964	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2011 Waterford 3		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**6**	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q96	Tier #		3
	Group #		
	K/A #	**2.2.7** Knowledge of the process for conducting special or infrequent tests.	
	Importance Rating		3.6

Proposed Question:

RS1737, "Post Refueling Low Power Physics Testing" is being performed.

The reactor is critical at 1x10-8 amps. Control Bank A (CBA) rod worth is being measured.

Who is required to approve Reactor Engineering reactivity recommendations?

A. Shift Manager

B. IPTE manager

C. Lead Test Coordinator

D. Reactor Engineering Manager

Proposed Answer: A

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(6). Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming , and determination of various internal and external reactivity effects. Specifically administrative requirements associated with low power physics testing process.

A. **Correct.** Per OS1005, Precaution 3.29, "The SM shall approve reactor engineering recommendations that affect reactivity, including activities such as low power physics testing."

B. Incorrect but plausible. The IPTE Manager does have a managing role in the low power physics testing process, however the SM is the individual required to approve RE reactivity recommendations.

C. Incorrect but plausible. The Lead Test Coordinator is the point of contact from the Reactor Engineering Department and would be providing the operators with reactivity management recommendations, however the SM is the individual required to approve those recommendations.

D. Incorrect but plausible. The Reactor Engineering Manager may review RE reactivity

recommendations prior to the commencement of testing, however the SM is the individual required to approve them.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	RS1737, Post Refueling Low Power Physics Testing (Rev 07)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:				(As available)
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	X		
Question History:	Last NRC Exam			
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)				
Question Cognitive Level:	Memory or Fundamental Knowledge		X	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41			
	55.43	**6**		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q97	Tier #		3
	Group #		
	K/A #	**2.2.21** Knowledge of pre- and post-maintenance operability requirements.	
	Importance Rating		4.1

Proposed Question:

The Plant is operating at 100% power.

While performing RC-OT004, "RCS Vent Path Block Valve Quarterly":

- RC-V-122, "PORV 456A Block Valve" CLOSES but WILL NOT OPEN.
- Maintenance finds a bad power supply breaker to the MOV, and replaces the entire breaker assembly at the MCC.
- ALL of their required work package instructions have been completed.
- The clearance has been lifted, RC-V-122 is ENERGIZED and CLOSED.
- RC-V-122 is ready for operations' post-maintenance testing.

For these conditions:

What MINIMUM post-maintenance testing will be REQUIRED to verify compliance with Technical Specification 3.4.4?

(For each of the below actions, assume all valve stroke times and indications are within acceptable limits)

A. Open RC-V-122, no other actions required.

B. Open RC-V-122; then close; then re-open.

C. Cycle the associated PORV through one complete cycle, then open RC-V-122.

D. Cycle the associated PORV through one complete cycle, then open RC-V-122; then Close; then re-open.

Proposed Answer:	B

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(2), Facility operating limitations in the technical specifications and their basis. The proposed question pertains to "application of required Tech. Spec. actions and surveillance requirements".

A. Incorrect but plausible. It is plausible that verification of valve operability may be accomplished by stroking the valve from its post maintenance CLOSED position to OPEN, as this would show that the valve does function and places it in its normally required OPEN position.

B. **Correct.** Per Tech. Spec. 3.4.4, RCS Relief Valves, Surveillance Requirement 4.4.4.2 "Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ACTION b. or c., in Specification 3.4.4." This surveillance requirement would be utilized to determine operability once the block valve maintenance activity is completed.

C. Incorrect but plausible. The action of "cycling through one complete cycle" does align with Surveillance Requirement 4.4.4.2, and operability of a PORV is associated with the ability to operate that valve manually with its control switch, however block valve RC-V-122 is the component required to be cycled. The PORV is not required to be cycled.

D. Incorrect but plausible. The second half of the distractor does describe the required cycling for RC-V-122. Additionally, the action of "cycling through one complete cycle" does align with Surveillance Requirement 4.4.4.2, and operability of a PORV is associated with the ability to operate that valve manually with its control switch, however block valve RC-V-122 is the component required to be cycled. The PORV is not required to be cycled.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.4.4, RCS Relief Valves (Rev 118)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:				(As available)
Question Source:	Bank #	X	98384	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 Beaver Valley		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q98	Tier #		3
	Group #		
	K/A #	**2.3.4** Knowledge of radiation exposure limits under normal or emergency conditions.	
	Importance Rating		3.7

Proposed Question:

Given the following conditions:

- A LOCA outside containment occurred at 0130.
- A Site Area Emergency was declared at 0140.
- The broken line was manually isolated locally, but the operator performing the task was injured and cannot leave the area on his own.
- Initial dose rate estimates are 110 R/hr gamma.
- The rescue time for a 2-man team is estimated to be 10 minutes with a maximum of 15 minutes.

Under these circumstances, a rescue attempt ____________ .

A. by risk-informed volunteers may proceed **ONLY** with Site Emergency Director authorization

B. is **NOT** allowed because whole body exposure would exceed emergency dose limits

C. may be made by qualified individuals selected and approved by the Radiological Controls Coordinator

D. may be made without special authorization since 10CFR20 exposure limits will **NOT** be exceeded

Proposed Answer:	A

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(4), Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions. Specifically the Emergency Dose limits that are allowed to perform lifesaving activities.

A. **Correct.** Given the conditions in the question stem, the rescue team will be performing a

"lifesaving activity". The dose for each member of the rescue team will be (110R/hr)(.25hr)=27.5 R. Per procedure ER-4.3, "Radiation Protection During Emergency Conditions", "Figure 2: Emergency Dose Limits", a person may receive a dose of >25R for the purpose of performing a lifesaving activity or protecting large populations. The dose is allowed "only on a voluntary basis to persons fully aware of the risks involved". This Emergency Dose Limit allowance requires STED or SED approval.

B. Incorrect but plausible. There is an allowable emergency dose limit of up to 25R for "lifesaving activities" when a lower dose is not practicable. This limit does not include the "voluntary basis" stipulation. This distractor is plausible if the student misinterprets the conditions in the question stem, or has false knowledge of the 25R dose limit criteria.

C. Incorrect but plausible. It is true that the rescue attempt may be performed, however the emergency dose extension must be approved by the STED or SED.

D. Incorrect but plausible. This distractor is plausible if the student misinterprets the conditions in the question stem, or has false knowledge of the emergency dose limit criteria.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ER 4.3, Radiation Protection During Emergency Conditions (Rev 30)

Proposed references to be provided to applicants during examination:				
Learning Objective:	L1525I15SR			(As available)
Question Source:	Bank #	X	100892	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2012 Point Beach		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**4**	
Comments:			

Seabrook Station 2015 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q99	Tier #		3
	Group #		
	K/A #	**2.4.8** Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	
	Importance Rating		4.5

Proposed Question:

Plant conditions:

- 100% power.
- A fire develops in the MCB.
- The fire causes the following conditions:
  - Spurious cycling of plant equipment
  - The crew will be forced to evacuate the Control Room

As the Unit Supervisor, which of the following describes your response?

A. Trip the reactor and enter E-0, "Reactor Trip or Safety Injection". Perform OS1200.00, "Response to Fire or Fire Alarm Actuation" in parallel until the Control Room is accessible.

B. Enter OS1200.00, "Response to Fire or Fire Alarm Actuation"; entering E-0, "Reactor Trip or Safety Injection" is not required.

C. Trip the reactor and perform the immediate actions of E-0, "Reactor Trip or Safety Injection" then transition to OS1200.00, "Response to Fire or Fire Alarm Actuation".

D. Enter OS1200.00, "Response to Fire or Fire Alarm Actuation"; entering E-0, "Reactor Trip or Safety Injection" is required after the Control Room is accessible.

Proposed Answer: B

Explanation (Optional):

SRO Justification:
This question meets SRO level screening criteria 10CFR55.43(b)(5), Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The proposed question requires "knowledge of the hierarchy, implementation, and/or coordination of plant normal, abnormal, and emergency procedures".

A. Incorrect but plausible. It is true that the reactor will be tripped, however OS1200.02, OAS item 2 states “”E-O should not be entered when the reactor is tripped in this procedure.” Additionally, the crew will transition from OS1200.00, “Response to Fire or Fire Alarm Actuation” to procedure 1200.02, “Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities”

B. **Correct**. Given the conditions in the question stem, the crew will be implementing procedure OS1200.02, “Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities”. Step 1 of OS1200.02 directs a manual reactor trip. OS1200.02, OAS item 2 states “”E-0 should not be entered when the reactor is tripped in this procedure.”

C. Incorrect but plausible. It is true that the reactor will be tripped, however OS1200.02, OAS item 2 states “E-0 should not be entered when the reactor is tripped in this procedure.” Additionally, the crew will transition from OS1200.00, “Response to Fire or Fire Alarm Actuation” to procedure OS1200.02, “Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities”

D. Incorrect but plausible. It is true that the operators will implement OS1200.00, however a control room evacuation is warranted, so OS1200.02 will be implemented. OS1200.02, OAS item 2 states “E-0 should not be entered when the reactor is tripped in this procedure.”

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1200.00, Response to Fire or Fire Alarm Actuation (Rev 22)			
Proposed references to be provided to applicants during examination:				NONE
Learning Objective:	L8210I07RO			(As available)
Question Source:	Bank #	X	100808	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2011 Point Beach		

(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NRC; failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	**5**	
Comments:			