Seabrook Station 2013 Written Exam ES-401.5

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q1	Tier #	1	
	Group #	1	
	K/A #	**007** Reactor Trip **EK3** Knowledge of the reasons for the following as they apply to a reactor trip: **EK3.01** Actions contained in EOP for reactor trip	
	Importance Rating	4.0	

Proposed Question:

E-0, "Reactor Trip or Safety Injection", requires Reactor Coolant Pumps to be tripped if RCS subcooling has decreased to <40°F with either a CCP or SI pump running.

What does this action prevent?

A. Damage to the RCP seal package due to the potential for a two-phase mixture existing in the pump casing.
B. Damage to the RCPs and RCS as a result of dynamic stresses associated with pumping a two-phase mixture.
C. Loss of the Unit Auxiliary Transformers as a result of potential RCP motor overloads or faults propagating back through the 13.8 kV buses.
D. Excessive loss of RCS water inventory through an RCS rupture which could lead to severe core uncovery if the RCPs were tripped later in the accident.

Proposed Answer: D

Explanation (Optional):

D is correct. Per Westinghouse Owners Group ERG's, Executive volume, RCP Trip/Restart the reason for purposely tripping the RCP' during accident conditions is to prevent excessive depletion of RCS water inventory through a small break in the RCS which might lead to severe core uncovery if the RCP's were tripped for some other reason later in the accident.

A is incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCP's is prudent. The document discusses tripping one RCP in procedure FR-C.2 to prevent pump damage due to running under the pumps under two-phase/voided conditions. This pump trip is to save that pump for potential future use. This situation is not applicable to the

guidance specific to the 40°F subcooling criteria in E-0.

B is incorrect but plausible. The Westinghouse background document discusses various situations where tripping the RCP's is prudent. The document discusses tripping one RCP in procedure FR-C.2 to prevent pump damage due to running under the pumps under two-phase/voided conditions. This pump trip is to save that pump for potential future use. This situation is not applicable to the guidance specific to the 40°F subcooling criteria in E-0.

C is incorrect but plausible. Per Westinghouse Owners Group ERG's, Executive volume, RCP Trip/Restart RCP operation with 2 phase mixture will affect pump performance, predominately pump current changes. It is plausible this could propagate back to the UATs through the 13.8 kv busses.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection (Rev 49) ERG Executive Volume, RCP Trip/Restart (Rev 2)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1202I 03 RO			(As available)
Question Source:	Bank #	X	TEB 16539	
	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	(CFR 41.5)	
	55.43		

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO
Q2	Tier #	1	
	Group #	1	
	K/A #	**008** Pressurizer (PZR) Vapor Space Accident (Relief Valve Stuck Open) **AK1** Knowledge of the operational implications of the following concepts as they apply to a Pressurizer Vapor Space Accident: **AK1.01** Thermodynamics and flow characteristics of open or leaking valves	
	Importance Rating	3.2	

Proposed Question:

The following plant conditions exist:

- A Pressurizer Power Operated Relief Valve (PORV) has failed open.
- The PORV cannot be closed or isolated.
- Safety Injection has been actuated.
- RCS Pressure is 1735 psig and slowly lowering.
- Pressurizer Relief Tank pressure is currently 10 psig.

How will the PORV Tailpipe Temperature Indicator trend over the next 30 minutes?

A. 240°F and stable.

B. Remain >400°F (Off-scale high).

C. 240°F and slowly decrease as RCS pressure decreases.

D. 240°F increasing then rapidly decrease to near 220°F then begin to slowly increase again.

Proposed Answer: D

Explanation (Optional):

D is correct. Isenthalpic expansion across the PORV from 1735 psig to the PRT pressure of 10 psig will give a tailpipe temperature of 240 °F. As the PRT pressure increases, the tailpipe temperature will increase. At 91 psig the PRT rupture disc will blow out and relieve the PRT to containment at near atmospheric pressure. This will lower the saturation temperature of the PRT and tailpipe temperature will lower. As containment pressure then rises the tailpipe temperature will rise.

A is incorrect but Plausible. Isenthalpic expansion across the PORV from 1735 psig to the PRT pressure of 10 psig will give a tailpipe temperature of 240 °F. If conditions do not change this

would be correct. But PRT pressure increases and PZR pressure decreases.

B is incorrect but plausible. Saturation temperature for 1735 psig is 617 °F. This temperature relieving from the PORV would cause the tailpipe temperature to peg high. Instrument range is 400 °F. This temperature will not be attained as the isenthalpic expansion will only get temperature to 240 °F.

C is incorrect but plausible. Isenthalpic expansion across the PORV from 1735 psig to the PRT pressure of 10 psig will give a tailpipe temperature of 240 °F. As the PZR pressure decreases the enthalpy of the saturated steam could be thought to decrease as well and the tailpipe temperature to decrease. Actually as the PZR pressure decreases the enthalpy of the saturated steam will increase. PRT temperature will increase and tailpipe temperature increases.

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 #			
	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	(CFR 41.8)	
	55.43		

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO
Q3	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: **EA2.36** Difference between overcooling and LOCA indications	
	Importance Rating	4.2	

Proposed Question:

Which of the following symptoms would most clearly distinguish a LOCA from a Steam Line break inside containment?

A. Increasing containment radiation levels.

B. Increasing containment sump levels.

C. Increasing containment pressure.

D. Decreasing pressurizer pressure.

Proposed Answer: A

Explanation (Optional):

A is correct. A LOCA inside containment would cause containment radiations levels to increase. A steam line break would not.

B is incorrect but plausible. Containment sump levels would increase for both a LOCA and a steam line break.

C is incorrect but plausible. Containment pressure would increase for both a LOCA and a steam line break.

D is incorrect but plausible. PZR level would decrease for both a LOCA and a steam line break.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection (Rev 49)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1203I01RO L1207I01RO			(As available)
Question Source:	Bank #	X	TEB14351	
	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	(CFR: 41.7)		
	55.43	(CFR 43.5 / 45.13)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q4	Tier #	1	
	Group #	1	
	K/A #	**011** Large Break LOCA **EK2** Knowledge of the interrelations between the and the following Large Break LOCA: **EK2.02** Pumps	
	Importance Rating	2.6*	

Proposed Question:

Plant conditions:

- A large break LOCA occurred.
- RCS pressure is 40 psig.
- All equipment functioned as designed.

Which of the following describes when the SI and RHR pumps <u>began</u> injecting into the RCS?

	SI Pumps (psig)	RHR Pumps (psig)
A.	1530	450
B.	1530	200
C.	1440	450
D.	1440	200

Proposed Answer: B

Explanation (Optional):

B is correct. 1530 psig is the shut off head for the SI pumps. 200 psig is the shut off head for the RHR pumps.

A is incorrect but plausible. 1530 psig is the shut off head for the SI pumps. 450 psig is the duscharge pressure when the RHR system is normally put in service.

C is incorrect but plausible. The SI pump will be injecting at 1440 psig but it began injecting at 1530 psig. 450 psig is the discharge pressure when the RHR system is normally put in service.

D is incorrect but plausible. The SI pump will be injecting at 1440 psig but it began injecting at 1530 psig. 200 psig is the shut off head for the RHR pumps.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	UFSAR, RHR Pump Curve, Fig 6.3-7 (Rev 15) UFSAR, SI Pump Curve, Fig 6.3-9 (Rev 15)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8034I07RO			(As available)
Question Source:	Bank #	X	TEB16320	
	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	(CFR 41.7)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q5	Tier #	1	
	Group #	1	
	K/A #	**015/017** Reactor Coolant Pump (RCP) Malfunctions **2.2.40** Ability to apply Technical Specifications for a system.	
	Importance Rating	3.4	

Proposed Question:

Plant conditions:

- RCS temperature is 557°F.
- All RCPs are operating.
- The Shutdown Banks are fully withdrawn.
- The Control Banks are fully inserted.
- "A" RCP vibrations go into HIGH alarm and the RCP is secured per the appropriate procedure.

What Technical Specification action must be taken first as a result?

A. No action required provided MODE is not changed.

B. Within 1 hour place the Control Rod Drive system in a condition incapable of rod withdrawal.

C. Return to all RCPs operating within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

D. No operations are permitted which could cause RCS dilution or core outlet temperature to be < 10°F below Saturation.

Proposed Answer: A

Explanation (Optional):

A is correct. T.S.3.4.1.2 requires at least two reactor coolant loops operable with two loops in operation when rod control is capable or rod withdrawal. Loss of one RCP leaves three loops operable and three loops operating with rod control capable of withdrawal. No action is required. The plant start up cannot proceed as Mode 2 requires all loops operable and operating.

B is incorrect but plausible. This action is associated with T.S. 3.4.1.2 action b. if only one loop is in operation.

C is incorrect but plausible. This action is associated with T.S. 3.4.1.2 action a. if less than two loops are operable.

D is incorrect but plausible. This action is associated with T.S. 3.4.1.2 action c. with no loops in operation.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.4.1.2 (Amendment No. 93)			
Proposed references to be provided to applicants during examination:				
Learning Objective:	L8021I09RO L1181I06RO			(As available)
Question Source:	Bank #	X	TEB27217	
	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: 41.10)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q6	Tier #	1	
	Group #	1	
	K/A #	**022** Loss of Reactor Coolant Makeup **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:

- Loss of all AC power.
- The crew is performing ECA-0.0, "Loss of All AC Power".
- NSOs have been dispatched to locally isolate RCP Seals.

Why are the RCP Seal Injection Throttle Valves closed in ECA-0.0?

A. Protect the Thermal Barrier Cooling system from steam formation.

B. Allow starting a charging pump without concern for damaging the RCPs or seals.

C. Ensure the necessary amount of ECCS flow is available, if required, when an emergency bus is energized.

D. Eliminate differential pressure across the Number 1 seals, allowing them to seat, thus minimizing RCS inventory loss.

Proposed Answer: B

Explanation (Optional):

B is correct. ECA-0.0 background for step 8 states seal injection is isolated to RCPs after prolonged loss of cooling to allow starting a charging pump without concern for thermally shocking RCPs.

A is incorrect but plausible. Thermal barrier steam formation is discussed and remedial actions taken in the ECA-0.0 Westinghouse background document. These actions are not required at Seabrook as the thermal barrier system is a separate closed loop system that is cooled by CCW.

C is incorrect but plausible. T.S. 3.4.6.2 Controlled leakage to RCP seals limit is to ensure in the event of a LOCA safety injection flow will not be less than assumed in the safety analyses.

D is incorrect but plausible. Previous revision of ECA-0.0 background discusses isolating seal return to allow seal return relief to lift and maintain 150 psig high pressure in the #1 seal leak off cavity to self limit #1 seal leak off flow.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ECA-0.0 Loss of All AC (Rev 42) ECA-0.0 Background (Rev 2) ECA-0.0 Deviation document (Rev 36)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8067I10RO			(As available)
Question Source:	Bank #	X	TEB20659	
	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	(CFR: 41.10)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q7	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: **AA1.02** Loads on the CCWS in the control room	
	Importance Rating	3.2	

Proposed Question:

While operating at 100% power the following trends are observed:

- C0768, Containment average temperature is 111°F and slowly increasing.
- A0285, RCP Thermal Barrier Inlet temperature is 94°F and slowly increasing.
- CS-TI-130, Letdown HX Outlet Temperature is 118°F and increasing.
- CS-TK-130, Letdown HX Temperature controller output is 100% and stable.

Which of the following could be the cause of these indications?

A. 1-CC-TK-2171, PCCW Loop "A" Supply header temperature controller output failing high.

B. 1-CC-TK-2171, PCCW Loop "A" Supply header temperature controller output failing low.

C. 1-CC-TK-2271, PCCW Loop "B" Supply header temperature controller output failing high.

D. 1-CC-TK-2271, PCCW Loop "B" Supply header temperature controller output failing low.

Proposed Answer: B

Explanation (Optional):

B is correct. "A" train PCCW temperature controller (CC-TK-2171) output failing low would cause TV-2171-1 (HX outlet) to close and CC-TV-2171-2 (HX bypass) to open. "A" train PCCW temperature would increase. Containment and RCP thermal barrier systems are cooled by both trains of PCCW. "A" train cooling water temperature increase would cause these temperatures to

increase. Letdown NRHX is cooled by "A" train of PCCW only. Question stem has letdown temperature increasing and the controller has increased to maximum trying to maintain it at setpoint.

A is incorrect but plausible. "A" train PCCW temperature controller (CC-TK-2171) output failing high would cause TV-2171-1 (HX outlet) to open and CC-TV-2171-2 (HX bypass) to close. "A" train PCCW temperature would decrease. This would result in temperature decrease of the supplied components. Question stem has temperatures increasing not decreasing.

C is incorrect but plausible. "B" train PCCW temperature controller (CC-TK-2271) output failing high would cause TV-2271-1 (HX outlet) to open and CC-TV-2271-2 (HX bypass) to close. "B" train PCCW temperature would decrease. This would result in Temperature decrease of the supplied components. Question stem has temperatures increasing not decreasing. Letdown NRHX is cooled by "A" train of PCCW only, changes to "B" train of PCCW would have no effect on letdown.

D is incorrect but plausible. "B" train PCCW temperature controller (CC-TK-2271) output failing low would cause TV-2271-1 (HX outlet) to close and CC-TV-2271-2 (HX bypass) to open. "B" train PCCW temperature would increase. Containment and RCP thermal barrier systems are cooled by both trains of PCCW. "B" train cooling water temperature increase would cause these temperatures to increase. Letdown is cooled by "A" train of PCCW and would be unaffected by this condition.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1212.01 PCCW System Abnormal (Rev13) OS1412.09 PCCW Monthly Flow Check (Rev8)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8036I04RO, 08RO			(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	(CFR 41.7)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q8	Tier #	1	
	Group #	1	
	K/A #	**027** Pressurizer Pressure Control System (PZR PCS) Malfunction **AA1** Ability to operate and / or monitor the following as they apply to the Pressurizer Pressure Control Malfunctions: **AA1.03** Pressure control when on a steam bubble	
	Importance Rating	3.6	

Proposed Question:

The following plant conditions exist:

- The unit is operating at 100%.
- PZR pressure is 2235 psig.
- PZR pressure control is in automatic.
- The "A" bank of backup heaters is energized in the "ON" position.
- The Master Pressurizer Pressure Controller malfunctions and the setpoint drifts from 2235 psig to 2160 psig, and components reposition.

After placing the Pressurizer Master Pressure Controller to MANUAL, what action will the Reactor Operator take with the Master Pressurizer Pressure Controller in response to the failure?

A. Operate the INCREASE pushbutton, which will close both spray valves and energize backup heaters "B", "C", and "D".

B. Operate the DECREASE pushbutton, which will close both spray valves and energize backup heaters "B", "C", and "D".

C. Operate the INCREASE pushbutton, which will de-energize backup heaters "B", "C", and "D", and open both spray valves.

D. Operate the DECREASE pushbutton, which will de-energize backup heaters "B", "C", and "D", and open both spray valves.

Proposed Answer: B

Explanation (Optional):

B is correct. When the controller setpoint drifts low an error is generated with the process signal being greater than the setpoint and the controller output will increase. The controller output will increase to a value that will demand the PZR control and back up heaters to de-energize and the spray valves to open. The "A" bank of back up heaters will not respond to the controller output with its control switch in "ON" and will remain energized. When PZR heaters de-energize and the spray valves open actual PZR pressure will decrease. Once the controller is in manual the operator will be required to raise PZR pressure. To do this the DECREASE pushbutton must be depressed to lower the controller output. This will cause the spray valves to close and the PZR heaters to energize. Again the "A" bank of heaters will not respond to controller output.

A is incorrect but plausible. Operation of the INCREASE push button will raise controller output and cause the spray valves to open and heaters to de-energize. This is the opposite of what should be done for the above description of events in the stem of the question. This is plausible as there is a need to raise PZR pressure and there is a common misconception with the operation of the master pressure controller. Increasing output acts to decrease pressure not increase pressure.

C is incorrect but plausible. Operation of the INCREASE push button will raise controller output and cause the spray valves to open and heaters to de-energize. This is the opposite of what should be done for the above description of events in the stem of the question. This is plausible as there is a need to raise PZR pressure and there is a common misconception with the operation of the master pressure controller. Increasing output acts to decrease pressure not increase pressure.

D is incorrect but plausible. Operation of the DECREASE pushbutton is the correct action but it will not cause the spray valves to open and the heaters to de-energize. This is plausible as there is a need to raise PZR pressure and there is a common misconception with the operation of the master pressure controller. Decreasing output acts to increase pressure by closing spray valves and energizing heaters.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Detailed System Text Fig 4.15 (Rev 6) 1-NHY-509026 Pressurizer Pressure Control Process Block Diagram (Rev9)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1182I04RO, L8027I06RO			(As available)
Question Source:	Bank #	X	TEB28613	
	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 41.7)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q9	Tier #	1	
	Group #	1	
	K/A #	**029** Anticipated Transient Without Scram (ATWS) **2.1.19** Ability to use plant computers to evaluate system or component status.	
	Importance Rating	3.9	

Proposed Question:

Plant conditions:

- The crew is performing the actions of FR-S.1, "Response to Nuclear Power Generation/ATWS" initiated by a Loss of All Feedwater event.
- The reactor is still critical.
- The crew is checking SG levels.
- All SG NR levels are off-scale low.
- All SG WR levels are approximately 62%.
- MPCS "C0722 - Total EFW Flow" is 750 gpm.

Based on these conditions, what action should the crew take to control EFW flow?

A. Throttle EFW flow to maintain RCS temperature.

B. Maintain current EFW flow until SG level is greater than 65% WR in at least two steam generators.

C. Maintain current EFW flow until SG level is greater than 15% NR in at least one steam generator.

D. Increase total EFW flow to greater than 880 gpm until SG level is greater than 6% NR in at least one steam generator.

Proposed Answer: D

Explanation (Optional):

D is correct. Continuous action step 8 of FR-S.1 requires at least one SG with NR level >6%. SG levels are all below the NR scale and FR-S.1 requires feeding at >880 gpm until one SG >6%NR. This is required to maintain secondary heat sink.

A is incorrect but plausible. Caution prior to step 8 of FR-S.1 cautions avoiding overfeeding SGs if SG levels are adequate to prevent reactivity additions from the cooldown. However SG levels are not adequate, heat sink is required to remove the heat generated from power operation and feeding at >880 gpm is required.

B is incorrect but plausible. This is criteria to maintain heat sink in E-1 not FR-S.1. C is incorrect but plausible. This is criteria to maintain heat sink in E-1 with containment adverse.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-S.1, Response to Nuclear Power Generation/ATWS			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1200I13RO			(As available)
Question Source:	Bank #	X	TEB23076	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2005 Seabrook NRC		
(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: 41.10)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q10	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: **EA1.27** Steam dump valve status lights and indicators	
	Importance Rating	3.9	

Proposed Question:

Plant conditions:

- A SG tube rupture has occurred.
- The crew is processing E-3, "Steam Generator Tube Rupture".
- RCS temperature is 548°F.
- Steam dumps are available and in the steam pressure mode.
- The BOP is currently cooling down to the target temperature of 495°F.
- Steam dump indications are as follows:
  - Steam dump demand is 30%.
  - Bank 1 Steam dump valve indications:
    - All Green indicating lights are ON and all Red indicating lights are ON.
  - Banks 2,3 and 4 Steam dump valve indications:
    - All Green indicating lights are ON and all Red indicating lights are OFF.

What are the implications of the steam dump indications?

A. Cooling down at the maximum rate. Steam dumps are operating correctly.

B. Not cooling down at the maximum rate. Bank 1 steam dumps are not operating correctly.

C. Not cooling down at the maximum rate. Bank 1 <u>and</u> 2 steam dumps are not operating correctly.

D. Not cooling down at the maximum rate. The BOP has not increased steam dump demand to the value required for maximum cooldown.

Proposed Answer:	B	

Explanation (Optional):

B is correct. E-3 step 7 initiates cooldown at maximum rate. With condenser available steam dumps are used. A steam dump demand signal of >25% should fully open all bank 1 dump valves. The indicating lights for steam dumps in this condition would be Bank 1 red lights on and green lights off and all other banks Green lights on and Red lights off. Indications given are Bank 1 valves in the intermediate position and not operating correctly.

A is incorrect but plausible. 30% demand is greater than maximum demand to get bank 1 steam dumps full open. Indications given in stem show Bank 1 steam dumps intermediate (not full open).

C is incorrect but plausible. 30% demand would fully open Bank 1 and partially open Bank 2 if Tave >550°F. RCS temperature < 550°F (P-12) isolates steam dumps. Bank 1 steam dump P-12 isolation can be bypassed and fully opened.

D is incorrect but plausible. Steam dump demand could be raised to 100% to open all steam dumps if Tave >550°F.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-3 SG Tube Rupture (Rev 42) Steam Dump detailed systems text. (Rev 5) 1-NHY-509050 MS Dump Control (Rev 5) 1-NHY-503662 MS Valve position lights (Rev 3)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8047I12RO, 13RO, 14RO			(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	(CFR 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q11	Tier #	1	
	Group #	1	
	K/A #	**040** Steam Line Rupture **AA2** Ability to determine and interpret the following as they apply to the Steam Line Rupture: **AA2.05** When ESFAS systems may be secured	
	Importance Rating	4.1	

Proposed Question:

Given the following:

- Reactor trip and Safety injection occurs from 100%.
- "B" SG is determined to be faulted outside of containment
- "B" MSIV is open and cannot be closed.
- "A","C" and "D" MSIVs are closed.
- The crew has transitioned to E-2, "Faulted Steam Generator Isolation", and is currently at step 7 "Check if ECCS flow should be reduced".
- Total available EFW flow is 400 gpm.
- "B" SG WR level is 10% and decreasing.
- "A" and "C" SG WR levels are 70% and increasing.
- "D" SG WR level is 60% and increasing.
- RCS temperature is 450°F and decreasing.
- RCS Pressure is 1450 psig and increasing.
- PZR level is 15% and increasing.

What is the next procedure transition?

A. Go to ES-1.1, "SI Termination".
B. Go to E-1, "Loss of Reactor or Secondary Coolant".
C. Go to FR-P.1, "Response to Imminent PTS Conditions".
D. Go to FR-H.1, "Response to Loss of Secondary Heat Sink".

Proposed Answer:	A	

Explanation (Optional):

A is correct. At step 7 of E-2 transition is made if subcooling is >40°F and total feed flow is >500 gpm or 2 WR SG levels >65% or 1 NR SG levels > 6% and RCS pressure stable or increasing and PZR level >7%. Subcooling is 143°F, 2 WR SG levels are 70%, RCS pressure is 1450 psig and increasing and PZR level is 15% and increasing.

B is incorrect but plausible. E-1 transition is made if the any above conditions were not met.

C is incorrect but plausible. Since E-0 has been exited critical safety functions apply and should be monitored. RCS temperature has decreased >100°F and this meets initial entry into the FR-P status tree. However the given RCS temperature does not meet requirements to go to FR-P.1.

D is incorrect but plausible. Since E-0 has been exited critical safety functions apply and should be monitored. Total available EFW flow is below 500 gpm which is the value required by the FR-H status tree to go to FR-H.1. However current SG levels are greater than required values to go to FR-H.1.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-2, Faulted Steam Generator Isolation (Rev 26) F-0.3, Heatsink (H) Tree (Rev 21) F-0.4, Integrity (P) Tree (Rev 21) Steam Tables

Proposed references to be provided to applicants during examination:				Steam Tables
Learning Objective:	L1207I02RO			(As available)
Question Source:	Bank #	X	TEB32376	
	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: 41.10)
	55.43	(CFR: 43.5 / 45.13)

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q12	Tier #	1	
	Group #	1	
	K/A #	**055** Loss of Offsite and Onsite Power (Station Blackout) **EK3** Knowledge of the reasons for the following responses as the apply to the Station Blackout: **EK3.01** Length of time for which battery capacity is designed	
	Importance Rating	2.7	

Proposed Question:

Which statement describes the Seabrook Station UFSAR Station Blackout coping evaluation?

A. Duration is 4 hours and relies only on the station batteries electrical power source.

B. Duration is 8 hours and relies on the station batteries and load shedding actions by operators.

C. Duration is 4 hours and relies on the Supplemental Emergency Power System (SEPS) as an alternate AC power source.

D. Duration is 8 hours and relies on the Supplemental Emergency Power System (SEPS) as an alternate AC power source.

Proposed Answer: A

Explanation (Optional):

A is correct. UFSAR Section 8.4 Station Blackout evaluation states the coping duration is four hours and the station batteries are the only source of electrical power.

B is incorrect but plausible. UFSAR section 8.3 gives the station batteries a rating of 8 hours and ECA-0.0 gives operator actions to shed DC loads to extend battery operations.

C is incorrect but plausible. The duration is 4 hours. UFSAR section 8.4.1 discusses that with the addition of SEPS it may be used for AC power BUT it will not be credited in the evaluation.

D is incorrect but plausible. UFSAR section 8.3 gives the station batteries a rating of 8 hours and UFSAR section 8.4.1 discusses that with the addition of SEPS it may be used for AC power BUT it will not be credited in the evaluation.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	UFSAR Section 8.4 Electrical power Compliance with 10CFR50.63, Loss of all AC (Station Blackout) (Rev 14) UFSAR Section 8.3 Electrical Power Onsite Power System pgs 68-70 (Rev 14) ECA-0.0, Loss of All AC Power (Rev 42)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1417I07RO			(As available)
Question Source:	Bank #	X	TEB28141	
	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	(CFR 41.5)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

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:

Given the following plant conditions:

- The plant was operating at 100% power when a Loss of Offsite Power occurred.
- All equipment functioned as designed.
- ES-0.1, "Reactor Trip Response" is being implemented.
- Offsite power will not be restored for two hours.
- The crew is verifying natural circulation.
- The following conditions exist:
  - SG Pressures: 1125 psig and stable.
  - PZR Pressure: 2200 psig and increasing slowly.
  - T-hot: 566°F in all four loops and stable.
  - T-cold: 563°F in all four loops and increasing slowly.
  - Core Exit TC and Subcooling indications are not available due to power loss to RVLIS/ICCM cabinets CP-486A and B.

What is the status of Natural Circulation, and what action is required, if any?

A. Not established. Increase dumping steam using ASDVs.
B. Not established. Increase dumping steam to the condenser.
C. Established. Heat removal is being maintained by the ASDVs. No action required.
D. Established. Heat removal is being maintained by the condenser steam dumps. No action required.

Proposed Answer:	A	

Explanation (Optional):

A is correct. Tcold is above Tsat for SG pressure and increasing. SG pressures are increasing. This indicates that natural circulation is not established. Action required per ES-0.1 step 10.b: "increase dumping steam". Must use ASDVs as the condenser is not available, loss of offsite power results in a loss of all CW pumps.

B is incorrect but plausible. Natural circulation is not established. Steam dumps would be used to dump steam if the condenser were available.

C is incorrect but plausible. Condition might be correct for natural circulation if analyzed incorrectly. But conditions met for natural circulation as stated above.

D is incorrect but plausible. Condition might be correct for natural circulation if analyzed incorrectly. But conditions met for natural circulation as stated above.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-0.1, Reactor Trip Response (Rev 37) Steam Tables

Proposed references to be provided to applicants during examination:				Steam Tables
Learning Objective:				(As available)
Question Source:	Bank #	X	TEB31641	
	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 41.10)
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q14	Tier #	1	
	Group #	1	
	K/A #	**058** Loss of DC Power **AA1**. Ability to operate and / or monitor the following as they apply to the Loss of DC Power: **AA1.01** Cross-tie of the affected dc bus with the alternate supply	
	Importance Rating	3.4*	

Proposed Question:

Plant conditions:

- 100% power.
- Normal electrical lineup.
- Battery charger 1B (BC-1B) DC circuit breaker fails open. 24 hours will be required to repair the breaker.
- DC bus 11B is being supplied from Battery 1B.
- Bus 11B voltage is 122VDC and lowering.
- Battery charger 1D (BC-1D), Battery 1D and DC bus 11D are operating normally at 132VDC.

Which of the following "B" Train 125VDC electrical lineups will allow continued 100% power operation until BC-1B is returned to service?

A. Bus 11B de-energized. Bus 11D supplied from Battery 11D and BC-1D

B. Bus 11B supplied from BC-1D and Battery 1D. Bus 11D supplied from BC-1D and Battery 1D.

C. Bus 11B supplied from BC-1D, Battery 1D and Battery 1B. Bus 11D supplied from BC-1D, Battery 1D and Battery 1B.

D. Bus 11B supplied from BC-2P and Battery 1B. Bus 11D supplied from BC-1D and Battery 1D.

Proposed Answer:	D

Explanation (Optional):

D is correct. OS1048.14 Vital Bus 11B Operation section 4.7 gives steps to place portable battery charger 1-EDE-BC-2P in service with the normal charger not in service. Portable battery charger 1-EDE-BC-2P is a replacement for the normal battery charger and has no T.S. time limit.

A is incorrect but plausible. Battery 11B would discharge until the bus is left de-energized with no operator action to place a charger in service. The plant will trip due to low SG level as the FRVs fail closed on a loss of bus 11B.

B is incorrect but plausible. Battery charger BC-1D can be cross connected to supply two battery buses. However in the conditions above Bus 11B would be de-energized in the switching operation to get to this cross connected configuration. This would result in a reactor trip from low SG levels as stated above. Also there is a 2 hour time limit to restore an operable battery charger to bus 11B.

C is incorrect but plausible. Breakers are available to electrically cross connect batteries. However there is a Kirk Key interlock system that prevents this.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1248.01 Loss of a Vital 125 VDC Bus (Rev 12) OS1048.14 Vital Bus 11B Operation (Rev 6) Detailed System Text Fig 3.2 (Rev 6)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8017I05RO, 06RO L1189I15RO			(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	(CFR 41.7)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q15	Tier #	1	
	Group #	1	
	K/A #	**Westinghouse E04** LOCA Outside Containment **EA1**. Ability to operate and / or monitor the following as they apply to the (LOCA Outside Containment) **EA1.2** Operating behavior characteristics of the facility.	
	Importance Rating	3.6	

Proposed Question:

Plant conditions:

- Reactor trip and SI.
- RDMS indicates high radiation levels in the RHR vaults.
- ECA-1.2, "LOCA Outside Containment" is in progress.
- RH-V14, "RHR Train A discharge to the RCS" and RH-V22, "RHR Train A cross-connect" have subsequently been closed.
- "A" train RHR and CBS pumps are in pull-to-lock.

The following conditions are observed:

- ECCS flow is decreasing.
- RCS pressure is 1100 psig and increasing.

Which of the following indicates the status of the LOCA and the subsequent procedural action that should be taken?

A. The LOCA is isolated. The crew should transition to ES-1.1, "SI Termination", step 1.

B. The LOCA is not isolated. The crew should continue with actions in ECA-1.2, "LOCA Outside Containment".

C. The LOCA is isolated. The crew should transition to E-1, "Loss of Reactor or Secondary Coolant" step 1.

D. The LOCA is not isolated. The crew should transition to ECA-1.1, "Loss of Emergency Coolant Recirculation" step 1.

Proposed Answer:	C

Explanation (Optional):

C is correct. Per ECA-1.2, step #4 if RCS pressure is increasing due to successful leak isolation the crew should transition to E-1.

D is incorrect but plausible. The crew would transition to ECA-1.1 at step #4 of ECA-1.2 if RCS pressure is not increasing due to leak isolation.

A is incorrect but plausible. A transition to ES-1.1 may ultimately be made to terminate SI based upon isolation of the LOCA , however this transition will most likely be made from E-1 not directly from ECA-1.2.

B is incorrect but plausible. If the initial actions in ECA-1.2 are not successful in isolating the LOCA additional actions may be taken to isolate valves in the other train.

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 #	X	TEB29959	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2007 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	(CFR: 41.7)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q16	Tier #	1	
	Group #	1	
	K/A #	**Westinghouse E11** Loss of Emergency Coolant Recirculation **EK2**. Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: **EK2.2** Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	
	Importance Rating	3.9	

Proposed Question:

Plant conditions:

- A Large Break LOCA is in progress:
- Cold leg recirculation has been established per ES-1.3, "Transfer to Cold Leg Recirculation".
- Containment pressure is 19 psig and decreasing.
- Just after returning to E-1, "Loss of Reactor or Secondary Coolant":
  - RH-P-8A is lost due to a broken shaft.
  - RH-P-8B trips on over-current.
- The crew enters ECA-1.1, "Loss of Emergency Coolant Recirculation".
- RWST level is 100,000 gallons.
- The crew is evaluating the ECA-1.1 CAUTION that states:
  - "If suction source is lost to any ECCS or spray pump, the pump should be stopped".

What pumps should be stopped?

A. Both SI pumps, only.
B. Both Charging pumps, only.
C. Both Charging pumps <u>and</u> both SI pumps, only.
D. Both CBS pumps, both SI pumps <u>and</u> both Charging pumps.

Proposed Answer:	C	

Explanation (Optional):

C is correct. After completing ES-1.3 the charging pump and SI pump suctions are supplied by RHR. With both RHR pumps lost there is no suction source for the charging or SI pumps.

A is incorrect but plausible. RHR is the suction source both SI pumps after completing ES-1.3. But they are not the only pumps supplied by RHR.

B is incorrect but plausible. RHR is the suction source both charging pumps after completing ES-1.3. But they are not the only pumps supplied by RHR.

D is incorrect but plausible. RHR is the suction source both SI and both charging pumps after completing ES-1.3. CBS pumps suction source is directly from the containment sump and does not require RHR pumps following ES-1.3.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ECA-1.1, Loss of Emergency Coolant Recirculation (Rev 36)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1209I03RO			(As available)
Question Source:	Bank #	X	TEB22246	
	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: 41.7)
	55.43	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q17	Tier #	1	
	Group #	1	
	K/A #	**Westinghouse E05** Loss of Secondary Heat Sink **EA2**. Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink) **EA2.2** Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	
	Importance Rating	3.7	

Proposed Question:

Plant Conditions:

- A loss of secondary heat sink has occurred.
- RCS bleed and feed has been established.
- All SG wide range levels are less than 5%.
- RCS hot leg temperatures on all loops are 560°F and increasing.
- The crew is preparing to reestablish EFW flow.

What EFW flow rate should be established, if any?

A. Feed one SG at the maximum rate.
B. Feed all SGs at 25 gpm until level is adequate.
C. Do not establish flow until the TSC completes evaluation.
D. Feed one SG at less than or equal to 100 gpm until level is adequate.

Proposed Answer: A

Explanation (Optional):

A is correct. FR-H.1 OAS page defines a "dry SG" as WR level <14%. The caution prior to step 21 gives guidance for feeding dry SGs. With feed and bleed established and SG WR levels at 5% and RCS temperature increasing then one intact SG should be fed at the maximum rate.

B is incorrect but plausible. There is a 25 gpm minimum feed requirement in ECA-2.1 for any SG with NR level < 6%. It is plausible the candidate could misapply the requirements of ECA-2.1 and incorrectly select this answer.

C is incorrect but plausible. The caution prior to step 21 states feedwater flow should not be established to more than one SG until TSC has evaluated refilling affected SGs. It is plausible the candidate could misapply this caution and incorrectly select this answer.

D is incorrect but plausible. The caution prior to step 21 states that if RCS temperatures are stable or decreasing then one SG is fed at 100 gpm. The question stem indicates RCS temperature is increasing.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-H.1, Response to Loss of Secondary Heat Sink (Rev 35) ECA-2.1, Uncontrolled Depressurization if all Steam Generators (Rev 34)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1211I02RO			(As available)
Question Source:	Bank #	X	TEB30069	
	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: 41.10)	
	55.43	(CFR: 43.5 / 45.13)	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q18	Tier #	1	
	Group #	1	
	K/A #	**077** Generator Voltage and Electric Grid Disturbances **AA2**. Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: **AA2.04** VARs outside the capability curve	
	Importance Rating	3.6	

Proposed Question:

Initial Main Generator/345KV Switchyard Conditions:

- Real Load: 1295 MWe
- Reactive Load: 150 MVARS out
- Switchyard Voltage: 358 KV
- Frequency: 60.0 Hz

The grid becomes unstable, and the BOP reports the following parameters:

- Switchyard Voltage has dropped to 348 KV.
- Frequency has remained at 60.0 Hz.

Assuming the reactor does NOT trip, how will Main Generator Amps respond to this event; and which limit (MWe or MVARs) will most likely be exceeded?

Generator Amps	Limit most likely to be exceeded
A. Increase	MVARs
B. Decrease	MVARs
C. Increase	MWe
D. Decrease	MWe

Proposed Answer: A

Explanation (Optional):

Since Frequency has not changed, turbine speed remains constant, since it is tied to the grid. The turbine control valves will remain steady, maintaining real load constant ("C" and "D" wrong). A generator's MVAR load increases when generator terminal voltage increases above grid voltage. This can be caused either by raising excitation voltage, or by decreasing grid voltage, which has happened in the transient described in the stem of the question. "A" is correct, and "B" wrong, since raising MVARs increases generator amps. "B", "C", and "D" are plausible, since a transient is in progress.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	General Physics Motors and Generators Text (Rev2) Figure 5-44			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1199I05RO			(As available)
Question Source:	Bank #	X		
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Millstone 2011 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: 41.5)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q19	Tier #	1	
	Group #	2	
	K/A #	**003** Dropped Control Rod **AA2**. Ability to determine and interpret the following as they apply to the Dropped Control Rod: **AA2.01** Rod position indication to actual rod position	
	Importance Rating	3.7	

Proposed Question:

Plant conditions:

- 75% power.
- Control bank "D" at 210 steps.
- A control bank "C" rod drops from 230 steps.
- The crew has entered OS1210.05, "Dropped Rod", and is recovering the dropped rod.

During rod recovery the NSO fails to hold the P/A converter Auto/Manual switch in the Manual position.

Which of the following describes the impact of this switch not being held in Manual?

A. The P/A converter will send improper rod height data to the RIL circuitry.
B. A "ROD CONTROL URGENT FAILURE" alarm will occur during the rod recovery.
C. The rod will be withdrawn to the correct height but with an incorrect DRPI indication.
D. C-11, "Auto Rod Out Motion Interlock", will actuate prematurely during the rod recovery.

Proposed Answer: A

Explanation (Optional):

A is correct. The P/A converter outputs are sent to comparison circuits for RIL alarms. OS1210.05 directs P/A converter Auto/Manual switch to be held in Manual. If the switch is left in Auto the P/A

converter will count step inputs from the rod control logic cabinet when CB C rod is recovered. With the actual bank position unchanged the RIL circuit would be in error.

B is incorrect but plausible. A 'Rod Control Urgent alarm" will be generated during the rod recovery. This alarm has nothing to do with the operation of the P/A converter.

C is incorrect but plausible. DRPI indications and Group step demand counters will respond to rod motion. But placement of P/A converter in Manual will not affect DRPI or GPI indications.

D is incorrect but plausible. C-11 Auto rod out motion interlock is generated from the P/A converter input. But it comes from CB D step inputs. The rod being recovered is CB C and thus would not affect inputs to C-11 if the Auto/Manual switch were left in Auto.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1210.05 Dropped Rod (Rev14) Rod Control Detailed Systems fig 3.12 and 3.13 (Rev3/05)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1185I02RO			(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	(CFR: 43.5)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q20	Tier #	1	
	Group #	2.	
	K/A #	**024** Emergency Boration **2.1.30** Ability to locate and operate components, including local controls.	
	Importance Rating	4.4	

Proposed Question:

Plant conditions:

- A fire has occurred in the control room and shutdown is being performed from the "A" train RSS panel.
- "B" train Essential switchgear is not accessible.
- CS-V-426, "Emergency Boration supply to Charging pump suction" is required to be opened locally.

Where is CS-V-426 located and how is it locally operated?

A. BAT room — Place AOV positioner in HAND, use hand wheel
B. BAT room — Declutch MOV, use hand wheel
C. VCT valve room — Place AOV positioner in HAND, use hand wheel
D. VCT valve room — Declutch MOV, use hand wheel

Proposed Answer: B

Explanation (Optional):

B is correct. CS-V-426 is a MOV located in the BAT room. MOV requires use of declutching lever to locally operate.

A is plausible but incorrect. CS-V-426 is in the BAT room but it is not an AOV. AOVs require positioner to be placed in hand and use of hand wheel for local operations.

C is plausible but incorrect. The VCT valve room contains charging pump suction isolation valves. It is plausible the Emergency Boration supply to the charging pump suction could be located in this room.

D is plausible but incorrect. The VCT valve room contains charging pump suction isolation valves. It is plausible the Emergency Boration supply to the charging pump suction could be located in this room.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	PID-1-CS-B20729 (Rev 17) PID-1-CS-B20725 (Rev 30) OS1200.02 Safe Shutdown and Cooldown from the RSS Facilities (Rev 17) OS1090.01 Manual Operation of Remote Operated Valves (Rev 15)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8025I18RO			(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	(CFR: 41.7)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q21	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: **AK2.03** Controllers and positioners	
	Importance Rating	2.6	

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- All Control Systems are operating in automatic.
- The backup pressurizer level control channel fails low.
- The Pressurizer Master Level Controller, RC-LK-459 and Charging Flow Controller, CS-FK-121 remain in AUTOMATIC.

How do RC-LK-459 and CS-FK-121 respond?

A. RC-LK-459 output increases. CS-FK-121 output increases.

B. RC-LK-459 output increases. CS-FK-121 output decreases.

C. RC-LK-459 output decreases. CS-FK-121 output increases.

D. RC-LK-459 output decreases. CS-FK-121 output decreases.

Proposed Answer: D

Explanation (Optional):

D is correct. When the backup level control channel fails low (<17%) letdown is isolated by RC-LCV-460. With letdown isolated actual PZR level increases. Actual PZR level greater than setpoint for the primary controller will cause its output to decrease. This decreasing output is an input to CS-FK-121. This decreasing input to CS-FK-121 will cause its output to decrease as well. The decrease in CS-FK-121 output will close CS-FK-121 and charging flow will be reduced to lower PZR level.

A, B and C are incorrect but plausible as they refer to the controllers output change. Increasing and decreasing controller outputs cause different responses in different systems depending if they are reverse acting or direct acting controllers

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509027, PZR Level Control Process Block Diagram (Rev10)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L8027I05RO				(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	(CFR 41.7)			
	55.43				
Comments:					

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q22	Tier #	1	
	Group #	2	
	K/A #	**037** Steam Generator (S/G) Tube Leak **AK3**. Knowledge of the reasons for the following responses as they apply to the Steam Generator Tube Leak: **AK3.08** Criteria for securing RCP	
	Importance Rating	4.1	

Proposed Question:

The following sequence of events occur:

- A 140 gpm SG Tube Leak occurred 3 hours ago.
- The reactor was tripped due to VCT level control criteria.
- ES-0.1, "Reactor Trip Response" and OS1227.02, "Steam Generator Tube Leak" are being performed in parallel.
- All RCP's are operating.
- Tave is 557°F and stable on the steam dumps.
- The RCS is being depressurized to 1500 psig to minimize subcooling.

Subsequently:

- The spray valves are closed late at 1450 psig.
- RCS pressure reduced to 1400 psig and turned.
- Subcooling now indicates +30°F and is slowly increasing.
- PZR level is 25% and stable.

What actions should be taken?

A. Stop all RCP's and actuate SI.
B. Stop all RCP's and restore subcooling.
C. Maintain all RCP's operating and actuate SI.
D. Maintain all RCP's operating and restore subcooling.

Proposed Answer: D

Explanation (Optional):
D is correct. A Note prior to step 1 states, "RCP subcooling trip criteria does not apply" in OS1227.02, Steam Generator Tube Leak abnormal and a Note prior to Step 15 discusses restoration of subcooling and wait to see if actions are successful prior to implementing SI actuation criteria of ES-0.1, Reactor Trip Response. A is incorrect but plausible as these are actions of E-0 for a loss of subcooling (<40°). However notes in OS1227.02, SG Tube Leak discuss not taking these actions. B is incorrect but plausible as stopping RCPs is required in E-0 on a loss of subcooling. However notes in OS1227.02, SG Tube Leak discuss not taking these actions. C is incorrect but plausible as loss of subcooling is SI actuation criteria in ES-0.1. However notes in OS1227.02, SG Tube Leak discuss not taking these actions.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1227.02, Steam Generator Tube Leak (Rev 19) ARG-3 SG Tube Leak Background, (Rev 2)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1190I02RO, 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	(CFR 41.5)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q23	Tier #	1	
	Group #	2	
	K/A #	**059** Accidental Liquid Radwaste Release **AK2.** Knowledge of the interrelations between the Accidental Liquid Radwaste Release and the following: **AK2.01** Radioactive-liquid monitors	
	Importance Rating	2.7	

Proposed Question:

The following plant conditions exist:

- A release from the "B" Waste Test Tank (WTT) is in progress at 45 gpm in accordance with ON1018.09, "Waste Test Tank 'B' Discharge to Transition Structure".
- RDMS channel R-6509-1, Liquid Waste Tank to CW System, goes into HIGH alarm.
- 1-WL-FCV-1458-1, high capacity waste distillate to discharge structure valve indicates as follows:
  - Red light - on
  - White light - on
  - Green light - on

What is the status of 1-WL-FCV-1458-1 and what actions are required?

A. 1-WL-FCV-1458-1 is closed. Recirc WTT and flush R-6509-1 until below alarm setpoint and restart discharge.

B. 1-WL-FCV-1458-1 is open. Close 1-WL-FCV-1458-1 from Waste Building control panel CP-38.

C. 1-WL-FCV-1458-1 is open. Close 1-WL-FCV-1458-1 from RDMS control consol CP-295.

D. 1-WL-FCV-1458-1 is closed. Notify chemistry to sample and check LEW permit.

Proposed Answer: B

Explanation (Optional):

B is correct. Valve indicating lights show the valve in mid position which is still an open position. White light indicates a high radiation trip has been actuated and locked in. OS1252.01 requires discharge path to be isolated, manually if required and chemistry to sample the affected system. 1-WL-FCV-1458-1 is required to be closed locally from CP-38.

A is incorrect but plausible. Given indications for FCV-1458-1 are open not closed. WTT recirc is required to sample WTT and there are steps in ON1018.07 to flush WTT discharge line if required. Discharge cannot be restarted until another sample is taken.

C is incorrect but plausible. During WTT discharge setup 1-WL-FCV-1458-1 is verified to trip close by inserting a false high rad trip from CP-295 in the Main Control Room. Current conditions in the stem of the question already indicate this high rad trip condition exists and the valve is not closed. 1-WL-FCV-1458-1 is required to be closed locally from CP-38.

D is incorrect but plausible. Given indications for FCV-1458-1 are open not closed. Chemistry will be required to sample and check/reissue another LEW.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1252.01 Process or Effluent High Radiation (Rev 16) ON1018.09 WTT B Discharge to Transition Structure(Rev 12) ON1018.07 WTT Recirculation (Rev 5)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8059I06RO L1187i02RO			(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	(CFR 41.7)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q24	Tier #	1	
	Group #	2	
	K/A #	**076** High Reactor Coolant Activity **AA2.** Ability to determine and interpret the following as they apply to the High Reactor Coolant Activity: **AA2.03** RCS radioactivity level meter	
	Importance Rating	2.5	

Proposed Question:

A review of chemistry parameters indicates the following trends:

- Gross activity (uCi/gm) – increasing
- RCS Dose Equivalent Iodine -131 (uCi/gm) – constant
- Letdown monitor (mr/hr) – increasing
- Secondary Dose Equivalent Iodine -131 (uCi/gm) – none detectable

Which of the following is a potential cause for these trends?

A. Normal power increase.
B. Fuel rod cladding failure.
C. Decrease in letdown temperature.
D. Inadvertent pH reduction of the RCS.

Proposed Answer: D

Explanation (Optional):

D is correct. Indications of RCS gross activity increasing and Iodine-131 stable is a potential crud burst. RCS pH reduction would cause a crud burst such as the peroxide addition at the beginning of refueling outages.

A is incorrect but plausible. It is plausible for a power increase to causing increasing trends in Dose Eq. Iodine. T.S 3.4.8 RCS Specific Activity requires sampling for I-131 within 2 to 6 hours following 15% power change in an hour. It is not expected to have a crud burst during a normal power increase (w/in 10%/hr) that would cause the letdown monitor indication to increase.

B is incorrect but plausible. A fuel clad failure would cause letdown monitor and gross activity to increase, but it would also cause Dose Eq. Iodine to increase as well.

C is incorrect but plausible. Ion exchange resins are very sensitive to elevated temperature. A temperature increase could cause a release of captured ions and an increase in radiation from the ion exchanger bed. The letdown monitor is upstream of the IX's and also the answer refers to letdown temperature decrease.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S. 3.4 8, Reactor Coolant System, Specific Activity (Amendment No. 115) OS1202.05, Reactor Coolant System High Activity (Rev 12)			
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			
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q25	Tier #	1	
	Group #	2	
	K/A #	**Westinghouse E13** Steam Generator Overpressure **EA1.** Ability to operate and / or monitor the following as they apply to the (Steam Generator Overpressure) **EA1.3** Desired operating results during abnormal and emergency situations.	
	Importance Rating	3.1	

Proposed Question:

Plant conditions:

- A spurious automatic "A" Train MSI occurred and cannot be reset.
- The reactor tripped.
- The crew has entered ES-0.1, "Reactor Trip Response".
- "A" SG exceeds 1225 psig and the crew enters FR-H.2, "Response to Steam Generator Overpressure".

The crew is currently attempting to dump steam from the "A" SG via the following paths per FR-H.2:

- "A" ASDV
- "A" MSIV Bypass valve
- MS-V-393 "A" SG supply to TDEFW.
- MSD-V-44 "A" MS upstream drain valve.

Under these conditions, which of these flow paths are available to be opened by the operators from the control room?

A. "A" ASDV and MS-V-393.

B. MSD-V-44 and MS-V-393.

C. MSD-V-44 and MSIV Bypass valve

D. MSIV Bypass valve and "A" ASDV.

Proposed Answer:	A	

Explanation (Optional):

A is correct. FR-H.2 step 4 gives four potential methods to dump steam from the affected SG. ASDVs, MSIV bypass valves, Steam supply valves to turbine driven EFW pump and MSIV upstream drains. With an "A" train Main Steam Isolation signal in that can not be reset then the MSIV bypass and MSIV upstream drains can not be opened from the control room. That leaves the ASDVs and the turbine driven EFW pump supply valve from the "A" SG.

B, C and D are plausible but incorrect. All answers contain flow paths listed in FR-H.2, but contain a valve blocked from opening due to the "A" train Main steam isolation.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-H.2, Response to SG Overpressure (Rev 19) 1-NHY-503669, MSIV Bypass valve logic (Rev 10) 1-NHY-503670, ASDV logic sht1 (Rev 11) 1-NHY-503695, Upstream Drain logic (Rev 2) 1-NHY-503585, MS-V-393 logic (Rev 13)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1211I06RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X		(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Modified Millstone 2009 Q26		

(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: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q26	Tier #	1	
	Group #	2	
	K/A #	**Westinghouse E15** Containment Flooding **EK2**. Knowledge of the interrelations between the (Containment Flooding) and the following: **EK2.2** Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	
	Importance Rating	2.7	

Proposed Question:

The following plant conditions exist:

- 100% power when a large break LOCA occurs.
- All plant systems respond as designed.
- The crew has entered FR-Z.2, "Response to Containment Flooding".
- The PSO has been directed to check the following penetrations isolated:
  - Reactor Makeup Water (RMW)
  - Primary Component Cooling Water (PCCW)
  - Fire Protection (FP)

How will the PSO check the penetrations isolated, and which penetration(s), if any, need to be isolated?

A. The containment isolation valves for all three systems can be checked on the UL panels. PCCW will need to be isolated.

B. The containment isolation valves for all three systems can be checked on the UL panels. All three paths should already be isolated.

C. The RMW and PCCW isolations can be checked on the UL panels. The FP path can only be verified locally and will require local operator action to isolate.

D. The RMW and PCCW isolation valves can be verified on the UL panels. The FP path can only be verified locally. All three paths should already be isolated.

Proposed Answer:	D	

Explanation (Optional):
D is correct. A large break LOCA will generate a "T" signal and a "P" signal. RMW-V-30 isolates on a "T" signal and indicates full closed on UL-3 "B" Train Phase A Isolation panel. PCCW isolation valves to containment isolate on "P" signal and indicate with red/green lights on the MCB. FP-V-592 is locked closed in modes 1,2,3 and 4 and has no remote indications of valve position. A is incorrect but plausible that all containment isolation valves indicate on the MCB. B is incorrect but plausible that all containment isolation valves indicate on the MCB. C is incorrect but plausible that Fire Protection equipment is always lined up to vital equipment areas.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-Z.2, Response to Containment Flooding (Rev 19) 1-NHY-503792, RMW V30 Logic Diagram (Rev 3) 1-NHY-503268, PCCW Isolation Valve Logic Diagram sh 1 of 2 (Rev 4) 1-NHY-503280, PCCW Isolation Valve Logic Diagram sh 2 of 2 (Rev 7) 1-NHY-503279, CC Monitoring Lights Logic Diagram (Rev 7) PID-1-FP-B20271, Fire Protection Details, Note 4 (Rev 21)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1212I10RO			(As available)
Question Source:	Bank #	X	TEB26915	
	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: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q27	Tier #	1	
	Group #	2	
	K/A #	**Westinghouse E03** LOCA Cooldown and Depressurization **2.4.3** Ability to identify post-accident instrumentation.	
	Importance Rating	3.7	

Proposed Question:

Plant conditions:

- Reactor Trip and SI has occurred due to a Small Break LOCA.
- Three control rods failed to insert.
- The crew is performing ES-1.2, "Post LOCA Cooldown and Depressurization".
- The Subcriticality Status Tree indicates an ORANGE path on SDS.

Which of the following instruments are to be used to validate the Orange Path?

A. Westinghouse IR NI's
B. Westinghouse SR NI's
C. Gamma-Metrics IR NI's
D. Gamma-Metrics SR NI's

Proposed Answer: C

Explanation (Optional):

C is correct. The Subcriticality Tree F-0.1 and station computer Critical Safety Function Status Tree use Post LOCA Gamma-Metric IR NI indications.

A is incorrect but plausible as Westinghouse IR NI's are checked in FR-S.1 for Reactor subcriticality.

B is incorrect but plausible as Westinghouse SR indications are used to declare the reactor critical during reactor startup.

C is incorrect but plausible as source range instruments are used to declare the reactor critical during reactor startup and Gamma-Metric does have SR indication.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	F-0.1, Subcriticality (S) Tree (Rev 20)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1200I15RO				(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	(CFR: 41.6)			
	55.43				
Comments:					

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q28	Tier #	2	
	Group #	1	
	K/A #	**003** Reactor Coolant Pump System (RCPS) **K2** Knowledge of bus power supplies to the following: **K2.01** RCPS	
	Importance Rating	3.1	

Proposed Question:

Bus voltage on 13.8 kV bus 1 begins to steadily decrease.

What components will automatically trip first?

A. "C" CW pump.

B. "A" and "B" RCPs.

C. "C" and "D" RCPs.

D. "A" and "B" CW pumps.

Proposed Answer: B

Explanation (Optional):

B is correct. A and B RCPs are powered from 13.8KV Bus 1. As bus voltage steadily decreases The RCPs trip at 70% nominal bus voltage after 1/3 second.

There is a common misconception as to the power supply to RCPs and CW pumps. These pumps are power by 13.8kv bus 1 and 2. The A and B RCPS and the A and C CW pumps are powered from Bus 1. The C and D RCPs and B CW pump is powered from bus 2.
The RCPs UV trip has a 1/3 second time delay. The CW pumps are stripped from the bus after a 1.5 second time delay.

A is incorrect but plausible as the "C" CW pump is powered from 13.8KV bus 1 and does get stripped from the bus on undervoltage.

C is incorrect but plausible due the common misconception of power source.

D is incorrect but plausible as CW pumps are stripped from the bus on under voltage and the common misconception as to power source.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-310004, 13.8kv swgr bus1-1, (Rev 16) 1-NHY-310882 sh.A05a, RCP 1-A Three Line Diagram, (Rev 10) 1-NHY-310882 sh.A05c,RCP 1-A Trip Schem. (Rev 7) 1-NHY-310101 sh.A03a, 13.8kv Bus 1 PT Three Line Diagram (Rev 6) 1-NHY-310101 sh.A03b, 13.8kv Bus 1 PTs schem Diagram (Rev 8) Reactor Coolant Detailed System Text page 46 RCP tripping (Rev 12/10) 13.8kv Detailed System text page 17 13.8kv Buss Stripping (Rev11/09)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8012I27RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	TEB2853	(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	(CFR: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q29	Tier #	2	
	Group #	1	
	K/A #	**004** Chemical and Volume Control System (CVCS) **K2** Knowledge of bus power supplies to the following: **K2.05** MOVs	
	Importance Rating	2.7	

Proposed Question:

Plant conditions:

- 100% power.
- Letdown flow is being controlled by CS-HCV-189.
- A loss of power panel PP-1E occurs.

What changes occur to letdown flow, if any and why?

A. Letdown increases. CS-HCV-189 fails open.
B. Letdown unaffected. CS-HCV-189 fails as is.
C. Letdown unaffected. CS-HCV-189 unaffected.
D. Letdown decreases. CS-HCV-189 fails closed.

Proposed Answer: D

Explanation (Optional):

D is correct. On a loss of PP-1E CS-HCV-189 fails closed. This will reduce letdown flow to zero.

A is incorrect but plausible. Failure of PP-1E does cause CS-HCV-182 Seal injection flow control valve to fail open.

B is incorrect but it is plausible that a loss of power to a MOV fails the valve as is.

C is incorrect but it is plausible that this is not a power supply to CS-HCV-189. Misconstrued with PP-1F and CS-HCV-190.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1247.02, Loss of Vital Instrument Bus PP 1E of PP 1F (Rev 15)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1186I 05 RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	TEB26408	(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: 41.7)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q30	Tier #	2	
	Group #	1	
	K/A #	**005** Residual Heat Removal System (RHRS) **K1** Knowledge of the physical connections and/or cause effect relationships between the RHRS and the following systems: **K1.04** CVCS	
	Importance Rating	2.9	

Proposed Question:

The following plant conditions exist:

- Mode 5.
- RCS temperature is 166°F.
- The pressurizer is water solid.
- Letdown Pressure Control Valve CS-PCV-131 is in automatic.
- RCS pressure is being maintained at 275 psig.
- An RHR control valve failure results in a rapid 10°F RCS temperature rise.
- Assume no operator action.

How does the Letdown system <u>initially</u> respond to the RCS heatup?

A. CS-PCV-131 OPENS. Letdown flow through RH-HCV-128 will DECREASE.
B. CS-PCV-131 CLOSES. Letdown flow through RH-HCV-128 will DECREASE.
C. CS-PCV-131 OPENS. Letdown flow through RH-HCV-128 will INCREASE.
D. CS-PCV-131 CLOSES. Letdown flow through RH-HCV-128 will INCREASE.

Proposed Answer:	C	

Explanation (Optional):

C is correct. Per OS1000.04, Plant Cooldown from Hot Standby to Cold Shutdown, with the plant in a solid condition RHR letdown flow is through CS-HCV-128 to CS-PCV-131. CS-PCV-131 is in automatic and the setpoint is adjusted to the desired pressure for the shutdown condition. While solid if the RCS heats up the expansion of water will increase RCS pressure. To maintain setpoint CS-PCV-131 will open to remove the additional volume of the RCS and flow through CS-HCV-128 will increase.

A is incorrect but is plausible if it the candidate assumes normal RHR letdown system operation where CS-HCV-128 controls letdown flow rate and CS-PCV-131 controls letdown system pressure. If RCS pressure were to increase then letdown flow would increase. CS-HCV-128 would be manually closed to maintain desired flow rate. While CS-PCV-131 would open to lower letdown pressure.

B is incorrect but plausible if the operation of CS-PCV-131 while in solid plant conditions is not understood. Normal letdown flow control valves while solid are fully open and an additional flow path exist if RCPs are operating. It is plausible that CS-PCV-131 closing could act to allow higher flow though normal flow path and reduce RCS pressure.

D is incorrect but plausible if it the candidate assumes normal RHR letdown system operation where CS-HCV-128 controls letdown flow rate and CS-PCV-131 controls letdown system pressure.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1000.04, Cooldown from Hot Standby to Cold Shutdown (Rev 40) OS1002.02, Operation of Letdown, Charging and Seal Injection Sect 4.4 (Rev 37) CVCS Detailed System Text Fig 3.10 (Rev 08/08) RHR Detailed System Text Fig 3.4 (Rev 3/05)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1166I07RO			(As available)
Question Source:	Bank #	X	TEB23140	
	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: 41.7)
	55.43	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q31	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: **K4.10** Control of RHR heat exchanger outlet flow	
	Importance Rating	3.1	

Proposed Question:

The following plant conditions exist:

- MODE 4 cooling down to MODE 5 for refueling outage.
- Train "A" Residual Heat Removal (RHR) is in service with flow set in automatic at 3500 gpm.
- The current cool down rate is 12°F/hr.
- The US orders the cool down rate increased to 40°F/hr.

Which of the following describes how the operator will increase the cool down rate and the associated response of the RHR system?

A. Throttles the RHR heat exchanger outlet valve in the closed direction such that RHR system water will spend more time in the RHR heat exchanger to be cooled further by PCCW. Flow is automatically increased through the RHR heat exchanger bypass line to maintain the combined flow rate constant at 3500 gpm.

B. Throttles more PCCW to the RHR heat exchanger such that RHR system water will be cooled further by PCCW. The RHR heat exchanger outlet and bypass valves require no throttling since RHR system flow rate remains constant at 3500 gpm.

C. Throttles the RHR heat exchanger bypass valve in the closed direction. This causes less water to flow through the RHR heat exchanger bypass line. Flow is automatically increased through the RHR heat exchanger to maintain the combined flow rate constant at 3500 gpm.

D. Throttles the RHR heat exchanger outlet valve in the open direction. This causes more RHR system water to flow through the RHR heat exchanger. Flow is automatically decreased through the RHR heat exchanger bypass line to maintain the combined flow rate constant at 3500 gpm.

Proposed Answer:	D	

Explanation (Optional):

D is correct. Opening the RHR heat exchanger outlet increases the flow rate through the heat exchanger. As flow rate increases through the heat exchanger the heat exchanger bypass valve would automatically close to maintain total system flow rate at the 3500 gpm setpoint. With more flow going through the heat exchanger cooldown rate would increase.

A is incorrect but plausible as closing the outlet valve of the heat exchanger would cause the water to spend more time in the heat exchanger. This will not increase the cooldown rate as more water bypasses the heat exchanger and cooldown rate would be reduced.

B is incorrect but plausible as this is performed late in the outage with very little or no heat load and leakage past flow control valves continues to cooldown the plant. This method is not used at the temperatures and cooldown rates specified in the stem of the question.

C is incorrect but plausible that closing the bypass valve would cause more flow to go through the heat exchanger and increase cooldown. However the outlet valve does not have an automatic control function it is manual only.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1013.03, Residual Heat Removal Train A Startup and Operation (Rev 26)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8033I 07 RO			(As available)
Question Source:	Bank #	X	TEB18608	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Seabrook 2009 NRC Remedial		

(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: 41.7)
	55.43	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q32	Tier #	2	
	Group #	1	
	K/A #	**006** Emergency Core Cooling System (ECCS) **A4** Ability to manually operate and/or monitor in the control room: **A4.07** ECCS pumps and valves	
	Importance Rating	4.4	

Proposed Question:

Plant conditions:

- Inadvertent "B" Train SI.
- Immediate actions are complete.
- PZR level 85% and increasing.
- RCS pressure is 2300 psig and stable.
- PSO is performing Attachment A and notes the following:
  - CS-P-2A and B operating
  - CS-P-2A miniflow valve CS-V-196 closed
  - CS-P-2B miniflow valve CS-V-197 open
  - Cold Leg Injection flow SI-FI-917 indicates 250 gpm
  - Charging header flow CS-FI-121 indicates 50 gpm
- Assume both charging pumps are operating evenly.

What actions should the PSO take with CS-V-196 /197, and why?

A. CS-V-196 should be opened. Low flow condition for pump exists. Miniflow required to cool pump.

B. CS-V-196 should be opened. SI signal and high flow condition should have automatically opened it.

C. CS-V-197 should be closed. High flow condition for pump exists. Miniflow should be closed to maintain NPSH.

D. CS-V-197 should be closed. SI signal and high flow condition should have automatically closed it.

Proposed Answer:	D	

Explanation (Optional):

D is correct. Using information given in stem total charging flow is approximately 360 gpm. (Cold leg injection = 250 gpm + normal charging header to seals = 50 gpm + CS-V-197 open recirc flow = 60 gpm = 360 gpm) With both pumps operating evenly then each pump is flowing forward 180 gpm. With an SI signal and high flow (122.5 gpm) CS-V-197 should have automatically closed. Attachment A of E-0 has a step to align cold leg injection by status panel. For the above conditions CS-V-197 should have closed automatically and the UL light should be on. With the valve open the UL light will be off and the operator should close CS-V-197.

A is incorrect but plausible as flow through the normal charging header is 50 gpm which is lower than the high flow setpoint.

B is incorrect but plausible as on an SI most valves open to inject flow. Indicated flow values are lower than those seen on small or large break LOCAs.

C is incorrect but it is plausible that the indications could be interpreted as too much flow, 360 gpm is greater than where charging pumps normally run, and closing CS-V-197 would be required.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection (Rev 49) CVCS Detailed System Text pg 14 (Rev 08/08) 1-NHY-503398, CS-Motor Operated Valves ESF Actuated Logic Diagram (Rev 6)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8024I05RO			(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	(CFR: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q33	Tier #	2	
	Group #	1	
	K/A #	**007** Pressurizer Relief Tank/Quench Tank System (PRTS) **K3** Knowledge of the effect that a loss or malfunction of the PRTS will have on the following: **K3.01** Containment	
	Importance Rating	3.3	

Proposed Question:

How is the Pressurizer Relief Tank (PRT) protected from overpressure <u>and</u> what Control Room indications alert the operator to its actuation?

A. Two rupture discs relieve pressure to Containment. PRT pressure will decrease and equalize with Containment pressure.

B. One rupture disc relieves pressure to Containment Sump "B". D5763, "Containment Sump 'B' Level High" alarm will actuate.

C. One relief valve and two rupture discs relieve pressure to Containment. PRT pressure will decrease and equalize with Containment pressure.

D. The pressure regulator in the nitrogen cover gas system will vent off excess pressure. D4472, "Pressurizer Relief Tank Pressure High" alarm will actuate.

Proposed Answer: A

Explanation (Optional):

A is correct. There are two rupture discs on the PRT with a setpoint of 91 psig that relieve to containment. The only indication the rupture discs have blown out is PRT pressure equalizing with containment.

B is incorrect but it is plausible there is only one rupture disc that is directed to the "B" containment sump.

C is incorrect but it is plausible there could be a relief valve on the PRT in addition to the rupture discs.

D is incorrect but it is plausible the nitrogen gas regulator that is designed to maintain 3 psig in the PRT could vent off excess pressure to maintain PRT pressure.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	PID-1-B20846, Reactor Coolant System Pressurizer (Rev 14)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8022I11RO			(As available)
Question Source:	Bank #	X	TEB31632	
	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	(CFR: 41.7)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q34	Tier #	2	
	Group #	1	
	K/A #	**007** Pressurizer Relief Tank/Quench Tank System (PRTS) **A3** Ability to monitor automatic operation of the PRTS, including: **A3.01** Components which discharge to the PRT	
	Importance Rating	2.7*	

Proposed Question:

Plant conditions:

- 100% power.
- Pressurizer Relief Tank (PRT) High Level and Pressure alarm is received.
- PRT level and pressure are increasing slowly.

Which of the following could cause these indications?

A. Letdown relief valve.
B. PORV packing leakoff.
C. RCP seal number 2 leakoff.
D. Alternate flow path for excess letdown.

Proposed Answer: A

Explanation (Optional):

A is correct the letdown relief valve inside containment discharges to the PRT.

B is incorrect but it is plausible that the PORV packing leakoff would be directed to the PRT since this is where the PORV discharge goes.

C is incorrect but it is plausible the RCP #2 seal leakoff goes to the PRT as it is a contaminated discharge path inside containment.

D is incorrect but it is plausible the excess alternate flow goes to the PRT as it is a contaminated discharge path inside containment.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	PID-1-B20846, Reactor Coolant System Pressurizer (Rev 14)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8022I14RO			(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	(CFR: 41.7)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q35	Tier #	2	
	Group #	1	
	K/A #	**008** Component Cooling Water System (CCWS) **A3** Ability to monitor automatic operation of the CCWS, including: **A3.01** Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS	
	Importance Rating	3.2*	

Proposed Question:

Plant conditions:

- 100% power
- CC-TV-2171-1, "A" PCCW HX outlet valve fails closed.
- OS1212.01, "PCCW System Malfunction" is being implemented.
- CC-P-11A is running.
- CC-P-11C is in standby.
- Train "A" PCCW Header temperature is now:
  - 135°F on CC-TI-2171 and increasing slowly.
  - 136°F on CC-TI-2197 and increasing slowly.

How do the Train "A" PCCW pumps respond?

A. CC-P-11A trips immediately. CC-P-11C starts after 30 seconds.

B. CC-P-11A trips after 60 seconds. CC-P-11C starts after 30 seconds.

C. CC-P-11A trips immediately. CC-P-11C does not automatically start. Neither pump can be started from the MCB.

D. CC-P-11A trips after 60 seconds. CC-P-11C does not automatically start. Neither pump can be started from the MCB.

Proposed Answer: D

Explanation (Optional):

Answer D is correct.

The following conditions will trip open a closed PCCW pump breaker:

High Temp: header temperature greater than 135°F as sensed on both CC-TI-2171 at

MCB, A0271 and CC-TI-2197 at CP-108A, A0296 for 60 seconds.
This interlock is only active if the RSS switch is in REMOTE.
If the running pump trips because of high temperature in the supply header, the alternate pump will not automatically start and cannot be started manually at the MCB.
This high temperature trip protects the pump and associated piping, which are restricted in their thermal expansion capabilities because of the seismic supports placed on them.

Answer A is incorrect but plausible.
The student could select this answer since the standby pump normally automatically starts 30 seconds after the running pump trips due to a fault. The student could forget that the high temperature interlock has a 60 second time delay and locks out both pumps

Answer B is incorrect but plausible.
The student could select this answer since the standby pump normally automatically starts 30 seconds after the running pump trips due to a fault. The student could forget that the high temperature interlock locks out both pumps.

Answer C is incorrect but plausible.
The student could select answer if he forgets that the high temperature interlock has a 60 second time delay.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503270, CC-PCCW Pumps SHT.1 Logic Diagram (Rev 9) PCCW Detailed System Text pg10 (Rev 11/03)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8036I07RO			(As available)
Question Source:	Bank #	X	TEB31639	
	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	(CFR: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

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: **K1.01** RPS	
	Importance Rating	3.9	

Proposed Question:

The following plant conditions exist:

- RCS pressure is 2235 psig,
- Reactor power is 8%,
- Both spray valves fail full open and remain open.

Assuming no operator action, which one of the following is most likely to be the cause of a reactor trip?

A. OP\DELTA-T.

B. OT\DELTA-T.

C. Low RCS pressure - SI at 1800 psig.

D. Low RCS pressure - reactor trip at 1945 psig.

Proposed Answer: C

Explanation (Optional):

C is correct. Failed open spray valves will rapidly lower RCS pressure. The reactor will trip at 1800 psig when an SI signal is generated. The LOW pressure Reactor trip (1945 psig) is blocked when below P7 (10%). OT\DELTA-T will not generate a trip at this low power. The pressure penalty will reduce the set point but not far enough to trip while at 8% power. RCS pressure does not affect the OP\DELTA-T setpoint.

A is incorrect but plausible as the RCS pressure input to OT\DELTA-T is easily confused with OP\DELTA-T.

B is incorrect but plausible since if at 100% power OT\DELTA-T setpoint is reduced far enough that it will generate a reactor trip before the LOW pressure trip at 1945 psig.

D is incorrect but plausible as this is the low pressure reactor trip setpoint and pressure is lowering rapidly.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-509046, RC PRZR Trip Signals W Functional Diagrams (Rev 7) Core Operating Limit Report pgs 1-3 (Rev 132) T.S. 2.2, Limiting Safety System Settings, Table 2.2-1 (Rev 101)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8027I06RO			(As available)
Question Source:	Bank #	X	TEB15666	
	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: 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: **K6.03** PZR sprays and heaters	
	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- 75% power.
- All control systems in automatic.
- Spray valve RC-PCV-455A fails open and cannot be closed by any means.
- The PSO reports pressure is 1950 psig and slowly decreasing.
- The US enters OS1201.06, "PZR Pressure Instrument/Component Failure".

Per OS1201.06, what action is required?

A. Energize all PZR heaters. Commence rapid down-power. Trip "C" RCP when less than P-8.

B. Commence rapid down-power. Raise charging flow to compress the PZR bubble. Trip "C" RCP when less than P-8.

C. Trip the reactor. Complete immediate actions of E-0, "Reactor Trip or Safety Injection". Trip "C" RCP and up to two more RCPs, as necessary.

D. Trip the reactor. Actuate SI. Enter E-0, "Reactor Trip or Safety Injection". Concurrently trip "C" RCP and up to two more RCPs, as necessary.

Proposed Answer: C

Explanation (Optional):

C is correct. OS1201.06 step 2 RNO directs reactor trip if pressure cannot be maintained >1945 PSIG. When immediate actions are complete then stop RCP supplying failed open spray valve. RC-PCV-455A is supplied by C RCP. If pressure continues to lower then stop a second and a third RCP.

A is incorrect but plausible. Energizing all pressurizer heaters would counteract the pressure decrease but not enough to compensate for a stuck open spray valve. It is also plausible to reduce

power below P8 so that an RCP can be stopped without requiring a reactor trip.

B is incorrect but plausible. Raising charging flow would have the effect of compressing the pressurizer steam space and would counteract the decrease in pressure but only to a marginal degree. Raising charging flow is not part of the strategy. It is also plausible that plant power would be reduced expeditiously and that the 'C' RCP would be secured when power is below P-8.

D is incorrect but plausible. It is true that the actions would include tripping the reactor and tripping the 'C' RCP and additional RCP's if necessary however the pressure decrease would not warrant a Safety Injection. Actuating Safety Injection is not part of the strategy.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1201.06, PZR Pressure Instrument/Component Failure (Rev 14)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1182I05RO			(As available)
Question Source:	Bank #	X	TEB26984	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Seabrook 2010 NRC		

(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: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

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: **A3.06** Trip logic	
	Importance Rating	3.7	

Proposed Question:

Plant startup is in progress with reactor power at 6%. Power range N-44 is out of service due to a failed detector and AOP actions are complete.

What is the condition of the P-10 Status Light and which of the following trips are enabled under these conditions?

A. Status light DEENERGIZED. Intermediate Range High Flux Reactor Trip.
B. Status light ENERGIZED. Intermediate Range High Flux Reactor Trip.
C. Status light DEENERGIZED. High PZR level Reactor Trip.
D. Status light ENERGIZED. High PZR level Reactor Trip.

Proposed Answer: A

Explanation (Optional):

A is correct. With PR N44 failed the abnormal procedure directs the affected channel bistables to be tripped by removing the control power fuses for N44. This trips associated bistables including P-10. The P-10 trip form N44 is one input of two required to actuate P-10 permissive. The P-10 status light will remain de-energized and the IR high flux trip will still be enabled.

B is incorrect but plausible as N44 is an input to P-10. The current status of P-10 indicating light is not in the question stem and IR high flux reactor trip is affected by the status of P-10.

C is incorrect but plausible as N44 inputs into P-10. The current status of P-10 indicating light is not in the question stem and high PZR level reactor trip is affected by the status of P-10.

D is incorrect but plausible as N44 is an input to P-10. The current status of P-10 indicating light is not in the question stem and high PZR level reactor trip is affected by the status of P-10.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1211.04, Power Range NI Instrument Failure (Rev 16) 1-NHY-509044, NI Perm and Blocks W Funtional Daigram (Rev 6) 1-NHY-509046, RC PRZR Trip Signals W Functional Diagrams (Rev 7)

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	(CFR: 41.7)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q39	Tier #	2	
	Group #	1	
	K/A #	**013** Engineered Safety Features Actuation System (ESFAS) **K3** Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: **K3.02** RCS	
	Importance Rating	4.3	

Proposed Question:

Given the following plant conditions:

- 100% power.
- The Reactor Operator reports the following valves have repositioned CLOSED.
  - CS-V149, "Letdown Line Containment Isolation".
  - CS-V167, "RCP Seal Water Return"
- All systems function as designed.

Based on these conditions, what procedure should be entered and what impact would this event cause if no operator action were to occur?

	Procedural Reference	Predicted Plant Impact
A.	OS1205.01, "Inadvertent Phase "A" Containment Isolation"	Automatic reactor trip on low PZR level.
B.	OS1205.01, "Inadvertent Phase "A" Containment Isolation"	Automatic reactor trip on high PZR level.
C.	OS1290.01, "Response to HELB Systems Actuation or Malfunction"	Automatic reactor trip on low PZR level.
D.	OS1290.01, "Response to HELB Systems Actuation or Malfunction"	Automatic reactor trip on high PZR level.

Proposed Answer: B

Explanation (Optional):

B is correct. Both CS-V-149 and CS-V-167 are closed by Phase "A" containment isolation. With

no operator action and letdown isolated the pressurizer would fill up and the reactor will trip on high PZR level (92%).

A is incorrect. But it is plausible as entry into OS1205.01 is the correct procedure. The reactor will not trip on low PZR level but it is plausible as low PZR level (7%) is a common set point for manual reactor trip such as in OS1201.02, RCS leak and OS1227.02, SG Tube Leak.

C is incorrect but plausible as CS-V-149 is isolated by HELB actuation as well as Phase A isolation. CS-V-167 is not isolated by HELB. The reactor will not trip on low PZR level but it is plausible as low PZR level (7%) is a common set point for manual reactor trip such as in OS1201.02, RCS leak and OS1227.02, SG Tube Leak.

D is incorrect but plausible as CS-V-149 is isolated by HELB actuation as well as Phase A isolation. CS-V-167 is not isolated by HELB. With no operator action and letdown isolated the pressurizer would fill up and the reactor will trip on high PZR level (92%).

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1205.01, "Inadvertent Phase "A" (Rev 15) OS1290.01, "Response to HELB Systems Actuation or Malfunction" (Rev 12)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1181I15RO			(As available)
Question Source:	Bank #	X	TEB32432	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2005 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	(CFR: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q40	Tier #	2	
	Group #	1	
	K/A #	**013** Engineered Safety Features Actuation System (ESFAS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the ESFAS; and (b) based Ability on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations; **A2.02** Excess steam demand	
	Importance Rating	4.3	

Proposed Question:

Plant conditions:

- RCS pressure has decreased to 1940 psig during a refueling outage cooldown.
- The "PRESS < P11 BLOCK PZR/MS SI" status light on UL-1 illuminates.
- In response, the crew performs the actions required by the MPE.
- As the cooldown continues, a steamline break occurs in the west pipe chase, on "A" steamline, upstream of its MSIV.

Assuming no operator action, what is the expected ESF response?

A. Both MSI and SI will occur regardless of break size.

B. MSI will always occur, but SI will only occur on a large break.

C. SI will always occur, but MSI will only occur on a large break.

D. Depending upon break size, MSI may occur, however SI will not occur.

Proposed Answer: D

Explanation (Optional):

When "PRESS < P11 BLOCK PZR/MS SI" status light on UL-1 illuminates OS1000.04, Plant Cooldown From Hot Standby to Cold Shutdown has the operator Block both trains of Pressurizer Safety Injection and both trains of Steamline Safety Injection. With these blocks established Low RCS pressure (1800#) and Low Steamline pressure (585#) will no

longer actuate Safety Injection. High containment pressure (HI-1 (4.3#)) and manual Safety Injection are still functional. Main Steam Isolation is now functional from steam line pressure rate of change (100# in 50 sec) and High containment pressure (HI-2 (4.3#)).

D is correct. The steam break occurs outside containment and upstream of the MSIVs. This will drop steam line pressure. Depending on size of break MSI may occur (100# in 50sec). A SI will not be generated as the low steam pressure (585#) is blocked and there will be no containment pressure increase as the break is outside of containment.

A is incorrect but plausible as a MSI and SI would be generated by Low RCS pressure from excessive cooldown or Low steam line pressure if not blocked.

B is incorrect but plausible as the MSI from high steam pressure rate of change is active in the current conditions and SI would occur if inside containment from HI-1.

C is incorrect but plausible as the MSI from high steam pressure rate of change is active in the current conditions and SI would occur if inside containment from HI-1.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1000.04, Plant Cooldown From Hot Standby to Cold Shutdown (Rev 40) 1-NHY-509046, RC PRZR Trip Signals W Functional Diagrams (Rev 7) 1-NHY-509047 sh.2, FW STM Gen Trip Signals W Functioanl Diagrams (Rev 1) 1-NHY-509048, Safeguards Actuation Signals W Functional Diagrams (Rev 17)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L80957I10RO			(As available)
Question Source:	Bank #	X	TEB32138	
	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: 41.5)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q41	Tier #	2	
	Group #	1	
	K/A #	**022** Containment Cooling System (CCS) **K4** Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: **K4.04** Cooling of control rod drive motors	
	Importance Rating	2.8	

Proposed Question:

The following plant conditions exist:

- Reactor Trip and Safety Injection actuated due to a large break LOCA.
- All safeguards systems are functioning as designed.
- Containment Pressure indicates 24 psig and slowly lowering.

What is the status of Containment Cooling systems?

	CRDM Cooling fans	Containment Recirculation fans
A.	Running	FILTER mode
B.	Tripped	FILTER mode
C.	Running	RECIRC mode
D.	Tripped	RECIRC mode

Proposed Answer: D

Explanation (Optional):

D is correct. At 18 psig in containment Hi-3 is actuated and a "P" signal is generated. The "P" signal trips CRDM cooling fans and automatically starts containment structure recirculation fans in RECIRC mode.

A is incorrect but plausible that the CRDM fans continue to operate as they cool the reactor vessel head during natural circulation cooldown and containment filter recirc could be needed in FILTER mode to clean up containment atmosphere following a large break LOCA.

B is incorrect but plausible as the CRDM cooling fans are tripped and containment filter recirc could be needed in FILTER mode to clean up containment atmosphere following a large break LOCA. C is incorrect but it is plausible that the CRDM fans continue to operate as they cool the reactor vessel head during natural circulation cooldown and the filter recirc fans do run in recirc mode.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503202, CAH-CRDM Cooling Fans Logic Diagram (Rev 7) 1-NHY-503204, CAH-Contn Structure Recirc Filter Fan Logic Diagram (Rev 5) 1-NHY-509048, Safeguards Actuation Signals W Functional Diagrams (Rev 17)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8038I04RO			(As available)
Question Source:	Bank #	X	TEB23133	
	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: 41.7)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q42	Tier #	2	
	Group #	1	
	K/A #	**022** Containment Cooling System (CCS) **2.1.20** Ability to interpret and execute procedure steps.	
	Importance Rating	4.6	

Proposed Question:

The PSO is shifting containment structure cooling units from CAH-F-1A, B, C, D and E to CAH-F-1B, C, D, E and F. When the shift is complete what will the position of CAH-F-1A control switch be and why?

A. PTL; Prevent starting following SI/LOP.

B. PTL; Prevent the fan from starting following LOP.

C. Normal; Allow start following LOP.

D. Normal; Allow auto start should one of the running units trip.

Proposed Answer: B

Explanation (Optional):

B is correct, Precaution 3.4 in OS1023.67, Containment Ventilation System Operation states the shutdown CAH fan is left in PTL so that it will not auto start on the DG during LOP. If it were in Normal after stop it would automatically start on the EPS at SR3.

A is incorrect but plausible as it is left in PTL which would prevent start following LOP however the EPS logic prevents start following LOP/SI and PTL would not be required to prevent start.

B is incorrect but is plausible that the CAH fans would be left in Normal so that they would start following LOP.

C is incorrect but it is plausible that the CAH fans would be left in Normal so that they would start following trip of a running fan.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1023.67, Containment Ventilation System Operation, Precaution 3.4 (Rev 11) 1-NHY-310931 Sh AC5b, Containment Structure CLG Fan 1-F-1A Schematic Diagram (Rev 6) 1-NHY-310931 Sh AC5c, Containment Structure CLG Fan 1-F-1A Legend & Sw Development (Rev 8)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8038I03RO, 04RO			(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	(CFR: 41.10)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q43	Tier #	2	
	Group #	1	
	K/A #	**026** Containment Spray System (CSS) **A4** Ability to manually operate and/or monitor in the control room: **A4.05** Containment spray reset switches	
	Importance Rating	3.5	

Proposed Question:

Plant conditions:

- Large Break LOCA inside containment.
- All equipment operated as designed.
- The crew is performing E-1, "Loss of Reactor or Secondary Coolant".
- RCS pressure 60 psig and stable.
- PZR level 0%.

Per E-1 when can the Containment Building Spray pumps be stopped, and which signal <u>must</u> be reset to place the pumps in standby?

A. Containment pressure less than 4 psig. P signal reset.
B. Containment pressure less than 4 psig. CBS signal reset.
C. Containment pressure less than 18 psig. P signal reset.
D. Containment pressure less than 18 psig. CBS signal reset.

Proposed Answer: B

Explanation (Optional):

B is correct. Per E-1 continuous action step 7, when containment pressure is less than 4 psig the CBS pumps are stopped and placed in standby. The CBS signal must be manually reset or the pumps will restart.

A is plausible but incorrect. Correct containment pressure and the P signal is reset in step 7 but the P signal does not input into the CBS pump start logic. The P signal is the Phase B containment isolation. The P and CBS signals are generated at the same time but have separate reset

switches.

C is plausible but incorrect. 18 psig is when CBS pumps are secured in ECA-1.1, Loss of Emergency Coolant Recirculation not in E-1.

D is plausible but incorrect. 18 psig is when CBS pumps are secured in ECA-1.1, Loss of Emergency Coolant Recirculation. Not in E-1.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-1 Loss of Reactor or Secondary Coolant (Rev41) 1-NHY-310900 shA61b (Rev7) 1-NHY-310900 shA61c (Rev6) 1-NHY-310900 shA61f (Rev11) 1-NHY-509048 (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	X	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(CFR: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q44	Tier #	2	
	Group #	1	
	K/A #	**039** Main and Reheat Steam System **K4** Knowledge of MRSS design feature(s) and/or interlock(s) which provide for the following: **K4.02** Utilization of T-ave. program control when steam dumping through atmospheric relief/dump valves, including T-ave. limits	
	Importance Rating	3.1	

Proposed Question:

Plant conditions:

- 100% power.
- An electrical grid disturbance causes a trip of the main turbine.
- A loss of offsite power occurs.
- All other plant systems and components respond as designed.
- Assume no operator action.

Where will reactor coolant temperature stabilize, and why?

A. 557°F, Condenser Steam Dump operation.

B. 557°F, Atmospheric Steam Dump operation.

C. 561°F, Atmospheric Steam Dump operation.

D. 567°F, Main Steam Safety Valve operation.

Proposed Answer: C

Explanation (Optional):

C is correct. The Main Steam Dumps are not available due to loss of offsite power. CW pumps are not running. RCS temperature will stabilize at 561°F as the ASDV's open at their 1125 psig setpoint. 561°F is associated with the saturation conditions @ 1125 psig.

A is incorrect but plausible as this is the temperature the condenser steam dumps will control at following a reactor trip if the main condenser were available.

B is incorrect but plausible as the ASDVs will be controlling temperature due to condenser not being available. ASDV setpoint is 1125 psig = 561°F saturation temperature. At step 1 of ES-0.1, Reactor Trip Response the setpoint of the ASDVs will be adjusted to control to 557°F. Prior to adjustment they will control at 561°F.

D is incorrect but plausible if neither condenser nor ASDVs were available the lowest MSSV setpoint = 1185 psig which corresponds to 567°F.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Steam Dump Detailed Systems Text Fig 4.2 and 4.5 (Rev 6/05) Main steam Detailed Systems Text page 8 MSSV Setpoint (Rev 4/08) Main steam Detailed Systems Text page 3 ASDV setpoint (Rev4/08)

Proposed references to be provided to applicants during examination:				
Learning Objective:	L804104RO, L8047I14RO			(As available)
Question Source:	Bank #	X	TEB30005	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Seabrook 2007		

(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: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q45	Tier #	2	
	Group #	1	
	K/A #	**039** Main and Reheat Steam System **K5** Knowledge of the operational implications of the following concepts as the apply to the MRSS: **K5.05** Bases for RCS cooldown limits	
	Importance Rating	2.7	

Proposed Question:

The following plant conditions exist:

- MODE 3.
- RCS Temperature is 375°F.
- "C" RCP is operating.
- Cool down is in progress with MS-PK-507 in AUTO in Steam Pressure Mode.
- OS1000.04, "Cooldown from Hot Standby to Cold Shutdown" limits the cool down rate to 100°F/Hr.

What is the basis for this limit?

A. Cool down rate in excess of 100°F/Hr will initiate a crack on the inner wall of the reactor vessel.

B. High cool down rates produce tensile stresses on the vessel inner wall which may exceed allowable stress.

C. High cool down rates produce tensile stresses on the vessel outer wall which may exceed allowable stress.

D. Cool down rate in excess of 100°F/Hr will cause exceeding the 320°F differential temperature limit between the PZR and charging.

Proposed Answer: B

Explanation (Optional):

B is correct. Basis for cool down rate limit is to prevent the tensile thermal stresses adding to total stress on the inner wall of the Reactor vessel and exceeding allowable RT_{NDT}. See basis for Tech.

Spec. 3.4.9

A is incorrect but plausible. Excessive stresses on the inner wall may propagate an existing flaw, not initiate a crack.

C is incorrect but plausible. Basis is tensile stresses on inner wall not the tensile stress on the outer wall.

D is incorrect but plausible. 320°F is limit between PZR vapor space and PZR spray fluid. This limit is to prevent thermal shock to the spray nozzle. With 'C' RCP in operation spray fluid is from RCS not charging.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S 3.4.9 Basis Reactor Coolant System Pressure/Temperature Limits (Rev BC 07-01)

Proposed references to be provided to applicants during examination:				
Learning Objective:	L1171I01RO			(As available)
Question Source:	Bank #	X	TEB30999	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	**2009 NRC Remediation 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	(CFR: 41.5)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q46	Tier #	2	
	Group #	1	
	K/A #	**059** Main Feedwater (MFW) System **K1** Knowledge of the physical connections and/or cause effect relationships between the MFW and the following systems: **K1.05** RCS	
	Importance Rating	3.1*	

Proposed Question:

Plant conditions:

- 100% power.
- Feedwater heater level transient resulted in HI-HI levels in the 25A and 26A feedwater heaters.
- Extraction steam to heaters 25A and 26A is isolated.
- Reactor does not trip.

What is the <u>initial</u> effect on the plant?

A. SG NR levels will increase. Actual reactor power will increase.
B. SG NR levels will increase. Actual reactor power will decrease.
C. SG NR levels will decrease. Actual reactor power will increase.
D. SG NR levels will decrease. Actual reactor power will decrease.

Proposed Answer: C

Explanation (Optional):

C is correct. Loss of feedwater preheating will add colder water to the SGs, reducing boiling, resulting in "shrink". The drop in Tcold adds positive reactivity, causing reactor power to increase.

A is incorrect but it is plausible the increase in reactor power would cause SG levels to swell.

B is incorrect but it is plausible the loss of extraction steam flow to the feedwater heaters would be a reduction in total steam flow and reactor power would decrease. D is incorrect but it is plausible the loss of extraction steam flow to the feedwater heaters would be a reduction in total steam flow and reactor power would decrease.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1290.02, Response to Secondary System Transient (Rev 11)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1191I08RO			(As available)
Question Source:	Bank #	X	TEB29987	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2007 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	(CFR: 41.5)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q47	Tier #	2	
	Group #	1	
	K/A #	**061** Auxiliary / Emergency Feedwater (AFW) System **K5** Knowledge of the operational implications of the following concepts as the apply to the AFW: **K5.02** Decay heat sources and magnitude	
	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- The Reactor tripped from 50% power, BOL.
- The crew has entered ES-0.1, "Reactor Trip Response".
- RCS temperature is 550°F and decreasing slowly.
- RCS pressure is stable at 2225 psig.

What could be the cause of the RCS temperature decrease?

A. Safety Injection has actuated.
B. Rupture of one of the steamlines.
C. High EFW flow coupled with low decay heat.
D. One of the turbine control valves has remained open.

Proposed Answer: C

Explanation (Optional):

C is correct. High EFW flow coupled with low decay heat will cause a slow RCS cooldown. Step 1 of ES-0.1, 'Reactor Trip Response' addresses RCS temperature control. The step includes the action to throttle EFW flow to maintain >500 gpm in the event that RCS temperature is less than 557°F and decreasing.

A is incorrect but plausible. A safety injection could cause an RCS cooldown, however there if there were a safety injection the crew would not have transitioned to ES-0.1, 'Reactor Trip Response'.

B is incorrect but plausible. A steamline rupture would cause an RCS cooldown, however the cooldown would be more severe and there would also be an associated loss of RCS pressure.

D is incorrect but plausible. A stuck open control valve could cause an RCS temperature decrease, however this would require a steam flowpath through the main steam isolation valves and turbine stop valves. E-0, 'Reactor Trip or Safety Injection' includes the immediate action to check the turbine tripped. The step specifically checks all stop or all control valves closed, and if not, then the operator is directed to close the main steam isolation valves.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-0.1 Reactor Trip Response (Rev 37)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L12251 04 RO			(As available)
Question Source:	Bank #	X	TEB30052	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 NRC Remediation 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: 41.5)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q48	Tier #	2	
	Group #	1	
	K/A #	**062** A.C. Electrical Distribution **K4** Knowledge of ac distribution system design feature(s) and/or interlock(s) which provide for the following: **K4.05** Paralleling of ac sources (synchroscope)	
	Importance Rating	2.7*	

Proposed Question:

Given the following conditions:

- 100% power.
- "A" EDG is started for surveillance testing.

When "A" EDG Output Breaker is closed onto Bus 5, how is a reverse power condition prevented?

A. Ensure synchroscope is rotating slowly in the 'FAST' direction.
B. Ensure synchronizing lights are out prior to closing the output breaker.
C. Ensure running and incoming voltages are matched prior to closing the output breaker.
D. Ensure running and incoming frequencies are matched prior to closing the output breaker.

Proposed Answer: A

Explanation (Optional):

A is correct. When paralleling AC sources the incoming source is set at a slightly higher frequency so that when the breaker is closed it will assume some real load and not be forced in a reverse power condition. This slightly higher speed is seen as the synchroscope rotating slowly in the FAST direction.

B is incorrect but plausible as this is a condition checked when closing the breaker. However this verifies sources are in sync when breaker is closed.

C is incorrect but plausible as this is done when paralleling AC sources. However this is to control reactive load not real load.

D is incorrect but plausible as frequencies are "approximately" matched. Incoming is still required to be higher, if exactly matched the syncroscope would not be rotating.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OX1426.01, DG 1A monthly Operability Surveillance (Rev 27)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8020I22RO			(As available)
Question Source:	Bank #	X		
	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	(CFR: 41.7)		
	55.43			
Comments:				

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ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q49	Tier #	2	
	Group #	1	
	K/A #	**063** D.C. Electrical Distribution **K3** Knowledge of the effect that a loss or malfunction of the DC electrical system will have on the following: **K3.02** Components using DC control power	
	Importance Rating	3.5	

Proposed Question:

Which of the following will result from a loss of VITAL 125 VDC buses 11A or 11B?

A. Loss of RVLIS/HELB plasma display.

B. Control room ventilation shifts to filter recirc mode.

C. Loss of associated train diesel generator auto start capability.

D. All condenser steam dump valves will be blocked from opening.

Proposed Answer: C

Explanation (Optional):

C is correct. Vital DC bus 11A and B supply associated train emergency diesel generator starting controls.

A is incorrect but plausible as the plasma displays for RVLIS/HELB are from vital AC power.

B is incorrect but it is plausible that this vital control system is powered from vital 125 DC..

D is incorrect but plausible as the solenoid valves for blocking condenser steam dumps coils are DC. However these coils are powered with individual rectifiers supplied from vital busses.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1248.01, Loss of a Vital 125 DC Bus (Rev 12) 1-NHY-350016, Bus Failure Analysis Vital 125 VDC (Rev 33)

Proposed references to be provided to applicants during examination:	None

Learning Objective:	L8020I23RO			(As available)
Question Source:	Bank #	X	TEB16019	
	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	(CRF: 41.7)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q50	Tier #	2	
	Group #	1	
	K/A #	**064** Emergency Diesel Generators (ED/G) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the ED/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.18** Consequences of premature opening of breaker under load	
	Importance Rating	2.6*	

Proposed Question:

Plant conditions:

- 100% power.
- EDG 'B' monthly surveillance is in progress with the DG at full load.
- The following alarms are received:
  - F7310, BUS E6 LOSS OF POWER.
  - D6352, BUS E6 UAT INC LNBKR TRIP & L/O.
- OS1246.02, "Degraded Vital AC Power" is entered.

Which of the following is the correct response to these conditions?

A. Restore power to bus using OS1246.02 Attachment "A", Restore power using Emergency Diesel.

B. Restore power to bus using OS1246.02 Attachment "C", Restore Emergency Bus power from offsite source.

C. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection", when immediate actions are complete trip RCP "A" and "D".

D. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection", when immediate actions are complete trip RCP "B" and "C".

Proposed Answer:	D

Explanation (Optional):
D is correct, With Bus E6 UAT breaker tripped and locked out a fault is indicated on Bus E6. With a fault in Bus E6 all breakers capable of supplying power Bus E6 are prevented from closing. Per OS1246.02 if power cannot be promptly restored to the bus then trip the reactor and when immediate actions are complete, secure the affected RCPs. Bus E6 supplies "B" train PCCW which supplies cooling to "B" and "C" RCPs. A is incorrect but plausible as no indications are given the DG experienced a fault. With the DG in parallel with offsite and the UAT breaker trips the DG breaker will trip open. The DG will remain running. B is incorrect but plausible as offsite power could be restored from the RAT if offsite were still available. C is incorrect but plausible as actions to trip reactor is correct but the wrong RCPs are designated to be tripped. It is a common misconception as to which RCPs are cooled by A or B train PCCW.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1246.02, Degraded Vital AC Power (Rev 13) D6352, Bus E6 UAT INC LN BKR Trip and L/O (Rev 03) F7310, Bus E6 Loss of Power (Rev 03)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1199I11RO			(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	(CFR: 41.5)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q51	Tier #	2	
	Group #	1	
	K/A #	**073** Process Radiation Monitoring (PRM) System **K1** Knowledge of the physical connections and/or cause effect relationships between the PRM system and the following systems: **K1.01** Those systems served by PRMs	
	Importance Rating	3.6	

Proposed Question:

The plant is at 75% power with the outlet of the Steam Generator Blowdown Flash Tank aligned to the condenser.

What automatic action occurs directly as a result of RM-6510, SG A Blowdown Sample Line Radiation Monitor exceeding the high alarm setpoint?

A. SB-V-1, SG "A" SG Blowdown IRC isolation valve closes.
B. Steam Generator Blowdown control valve SB-CV-6519 closes.
C. All Steam Generator Blowdown system inside containment isolation valves close.
D. All Steam Generator Blowdown system outside containment isolation valves close.

Proposed Answer: B

Explanation (Optional):

B is correct. Any of the Steam Generator sample line rad. monitors will send a CLOSE signal to SB-CV-6519.

A is incorrect but it is plausible the 'A' Steam Generator blowdown radiation monitor RM-6510 would close its associated containment blowdown isolation valve SB-V-1.

C is incorrect but plausible. The Inside Containment Isolation valves would eventually go closed however this is an indirect effect as they receive a CLOSE signal from a blowdown flash tank high level alarm.

D is incorrect but plausible. The Inside Containment Isolation valves would eventually go closed as

an indirect effect from a blowdown flash tank high level alarm however the Outside Containment Isolation Valves do not receive such a signal.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1252.01, Process or Effluent High Radiation (Rev 16) OS1227.01, Recovery From Steam Generator Blowdown System Isolation (Rev 12)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8059I06RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	TEB7582	(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 NRC Remediation		

(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	(CFR: 41.7)	
	55.43		

Comments:

Question as given in 2009 Remediation Exam:

Given the following plant conditions:

- The plant is at 75% power.
- The Steam Generator Blowdown Flash Tank is aligned to the ocean.
- RM-6510, SG A Blowdown Sample Line Radiation Monitor goes into HIGH ALARM.

What automatic action occurs DIRECTLY as a result of this condition?

A. None. This radiation monitor provides no automatic function.

B. SB-CV-6519, Steam Generator Blowdown Flash Tank Outlet Valve will CLOSE.

C. The Steam Generator Blowdown Inside Containment Isolation Valves will CLOSE.

D. The Steam Generator Blowdown Outside Containment Isolation Valves will CLOSE.

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Examination Outline Cross-reference:	Level	RO	SRO
Q52	Tier #	2	
	Group #	1	
	K/A #	**073** Process Radiation Monitoring (PRM) System **K5** Knowledge of the operational implications as they apply to concepts as they apply to the PRM system: **K5.01** Radiation theory, including sources, types, units, and effects	
	Importance Rating	2.5	

Proposed Question:

Primary to Secondary leak rate monitor is trending up due to a small fuel element failure. Which of the following Process Radiation Monitors inputs to the Primary to Secondary leak rate monitor and what type of radiation is causing the change in trend?

A. RCS Letdown Monitor RM6520-1 due to Gamma radiation.

B. "A" SG Blowdown Monitor RM6510-1 due to Beta radiation.

C. "A" Main Steamline Monitor RM6481-1 due to Gamma radiation.

D. Condenser Air Evacuation Monitor RM6505-1 due to Beta radiation.

Proposed Answer: D

Explanation (Optional):

D is correct. The primary to secondary leak rate monitor gets an input from RM-6505 Condenser Air Evac monitor. RM6505 uses a RD-25 Beta Scintillation detector.

A is incorrect but plausible. To give an accurate indication of Primary to Secondary leakage the chemistry department adjusts the conversion factor for the CPM detected by RM-6505 to GPD leakage. (Example: higher RCS activity with the same size leak would give more CPM detected.) It is plausible the letdown radiation monitor could input into the Primary to Secondary leak rate monitor to perform this function. RM-6520 uses a RD-10B GM detector which does detect gamma radiation.

B is incorrect but plausible. SG blowdown monitors do not input into the primary to secondary leak rate monitors. SG blowdown monitors do give indications of SG tube leakage. RM-6510 uses a RD-53 Gamma scintillation detector which detects gamma radiation not beta radiation.

C is incorrect but plausible. Main Steam line radiation monitors do not input into the primary to secondary leak rate monitor. Main Steam line radiation monitors do give indication for SG tube leakage. RM-6548-1 uses a RD-12 GM detector which does detect gamma radiation.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	CS0905.08 Response to a Primary to Secondary Leak (Rev06) OS1227.02 Steam Generator Tube Leak (Rev19) Detailed Systems excerpt for RD-25 detector. (Rev02/12)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8059I16RO			(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	(CFR: 41.5)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q53	Tier #	2	
	Group #	1	
	K/A #	**076** Service Water System (SWS) **A1** Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the SWS controls including: **A1.02** Reactor and turbine building closed cooling water temperatures	
	Importance Rating	2.6*	

Proposed Question:

Plant conditions:

- 100% power
- TA has occurred in both trains.
- SW-V-20, SW Train "A" to DTS Isolation, failed to close and cannot be closed locally.
- SW-P-110A, Cooling Tower pump has just been stopped.

Which of the following describes the effect of these conditions on SCCW and "A" train PCCW?

	SCCW temp	Train "A"PCCW temp
A	Increase	Increase
B	Increase	Stable
C	Stable	Stable
D	Stable	Increase

Proposed Answer: A

Explanation (Optional):

A is correct. When both trains of SW receive a Tower Actuation signal SW-V-4 and SW-V-5 close and all SW cooling to SCCW is isolated. With no cooling, SCCW temperatures will increase. Response to SW-V-20 failure to close requires shut down of SW-P-110A to prevent pumping the cooling tower water volume to the ocean and rendering both trains of SW inoperable. With no SW pumps running in the "A" train of SW the "A" train PCCW temperature will increase.

B is incorrect but plausible as SCCW temperature will increase and if not realized that stopping SW-P-110A secures all SW flow in "A" train of PCCW it would remain stable.

C is incorrect but plausible if not realized both trains of TA will close SW-V4 and SW-V-5 and isolate all SW to SCCW temperature will remain stable. If not realized that stopping SW-P-110A secures all SW flow in "A" train of PCCW it would remain stable.

D is incorrect but plausible as PCCW temperature will increase and if not realized both trains of TA will close SW-V4 and SW-V-5 and isolate all SW to SCCW temperature will remain stable.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1216.01, Degraded Ultimate Heat Sink (Rev 22)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1193I02RO L8037I13RO				(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	(CFR: 41.5)			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q54	Tier #	2	
	Group #	1	
	K/A #	**078** Instrument Air System (IAS) **A4** Ability to manually operate and/or monitor in the control room: **A4.01** Pressure gauges	
	Importance Rating	3.1	

Proposed Question:

Plant conditions:

- 100% power.
- The Service Air system had a leak from a failed air hose in use by maintenance.
- ON1242.01, "Loss of Instrument Air" is being performed.
- Pressure dropped to 84 psig before NSO's located and isolated the leak.
- Service Air Isolation Valves, SA-V-92 and SA-V-93 automatically closed.
- IA dryer outlet pressure indicators IA-PI-8015 and IA-PI-8010 now indicate 98 psig and increasing.

How is the service air header to be returned to service, if at all?

A. Locally reset and open SA-V-92/SA-V-93.

B. Cannot be restored at this time. IA pressure is not high enough.

C. Maintain IA header pressure and cycle SA-V-92/SA-V-93 MCB control switch open to restore service air headers.

D. Hold SA-V-92 control switch on MCB in open valve full open and then hold SA-V-93 control switch on MCB in open until valves are full open.

Proposed Answer: C

Explanation (Optional):

C is correct. Step 9 of OS1242.01 directs to maintain IA dryer outlet >95 psig and cycle openSA-V-92/93 to repressurize the service air header. The intent is to cycle open and closed SA-V-92/93 to slowly restore pressure to service air without dropping Instrument air <95 psig. SA-V-92/93 operate off one control switch and will open even if < 90 psig with the control switch held in the open position.

A is incorrect but plausible as there are local actions directed if cycling SA-V-92/93 cannot restore Service air header pressure.

B is incorrect but plausible if the reset values of the low pressure switch or pressure specified in the abnormal procedure are unknown.

D is incorrect but plausible. If the control switch for SA-V-92/93 were held in the open position SA-V-92/93 would fully open regardless of results to Instrument Air header pressure.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1242.01, Loss of Instrument Air (Rev 13) 1-HY-310863 sh E46/8a, Service Air Isolation Valves I-V-92 & I-V-93 Schematic Diagram (Rev 8)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1194I02RO			(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	(CFR: 41.7)	
	55.43		

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO
Q55	Tier #	2	
	Group #	1	
	K/A #	**103** Containment System **K4** Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: **K4.06** Containment isolation system	
	Importance Rating	3.1	

Proposed Question:

The plant is in MODE 5. Containment Pre-entry purge is in progress.

What is the response of the Containment Purge system if a Train "A" Containment Ventilation Isolation (CVI) signal is generated?

A. Containment Pre-entry Purge Supply fan (FN-9) trips, Containment Pre-entry Purge Exhaust fan (FN-10) trips, supply and exhaust penetrations isolate.

B. Containment Pre-entry Purge Supply fan (FN-9) trips, Containment Pre-entry Purge Exhaust fan (FN-10) trips, only supply penetration isolates.

C. Containment Pre-entry Purge Supply fan (FN-9) trips, supply and exhaust penetrations isolate.

D. Containment Pre-entry Purge Exhaust fan (FN-10) trips, only exhaust penetration isolates.

Proposed Answer: C

Explanation (Optional):

C is correct. An "A" train CVI signal trips CAP FN-9 and closes CAP-V-1 and CAP-V-4 supply and return valves ORC. CAP FN-10 does not trip.

A is incorrect but it is plausible CAP FN-10 would trip as it has no suction path When CAP-V-4 closes.

B is incorrect but it is plausible CAP FN-10 would trip as it has no suction path and only supply penetrations close as the discharge path is filtered.

D is incorrect but it is plausible that only the containment discharge fan and exhaust penetrations are secured to stop the containment release.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	1-NHY-503221, Cap Containment Air Purge Isolation Vavles Logic Diagram (Rev 8) 1-NHY-503222, CAP Containment Air Pre-entry & Refueling Supply Fans Logic Diagram (Rev 5) 1-NHY-530224, CAP Containment Air Purge Exhausst Fan FN-10Logic Diagram (Rev 7)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8038I24RO			(As available)
Question Source:	Bank #	X	TEB23149	
	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	(CFR: 41.7)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q56	Tier #	2	
	Group #	2	
	K/A #	**016** Non-Nuclear Instrumentation System (NNIS) **K1** Knowledge of the physical connections and/or cause effect relationships between the NNIS and the following systems: **K1.10** CCS	
	Importance Rating	3.1*	

Proposed Question:

Plant conditions:

- A LOCA has occurred.
- Actual Reactor Vessel level and void fraction are stable.
- Containment temperature is 201°F and increasing.

How is RVLIS Full and Dynamic range level indication affected by the increasing Containment temperature?

A. Both increase.

B. Both decrease.

C. Neither changes significantly.

D. Full range increases. Dynamic range decreases.

Proposed Answer: C

Explanation (Optional):

C is correct. Measurements of the sensing line temperatures provide the accuracy required for reactor vessel level measurement. These measurements, together with the core exit thermocouples and wide range pressure measurements, are employed by the RVLIS microprocessor to compensate the DP transmitters for differences between RCS density and reference leg density, particularly during the environmental changes inside containment following an accident.

A is incorrect but it is plausible that temperature changes to D/P sensing lines affect level indications.

B is incorrect but it is plausible that temperature changes to D/P sensing lines affect level indications. D is incorrect but it is plausible that temperature changes to D/P sensing lines affect level indications. For Dynamic and Full range levels indications differently.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	AMI Detailed System Text fig 3.4 (Rev 9/09) AMI Detailed System Text pg 5-7 (Rev 9/09)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8058I09RO			(As available)
Question Source:	Bank #	X	TEB18822	
	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	(CFR: 41.7)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q57	Tier #	2	
	Group #	2	
	K/A #	**033** Spent Fuel Pool Cooling System (SFPCS) **A1** Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with Spent Fuel Pool Cooling System operating the controls including: **A1.01** Spent fuel pool water level	
	Importance Rating	2.7	

Proposed Question:

The following plant conditions exist:

- The plant is in MODE 1.
- Spent fuel pool water level LO alarms are received in the control room.
- Control room operators enter OS1215.07, "Loss of Spent Fuel Pool Cooling" as directed by alarm response procedures.
- Spent Fuel Pool Level is 24.5 feet and rapidly decreasing.

Based on these conditions, which of the following describes the required source of makeup to the Spent Fuel Pool?

A. Reactor Makeup Water (RMW).
B. Refueling Water Storage Tank (RWST).
C. Chemical Volume Control System (CVCS).
D. Demineralized Water Storage Tank (DWST).

Proposed Answer: B

Explanation (Optional):

B is correct. Per OS1215.07 if SFP level is < 25.4 feet and decreasing rapidly direction is to commence emergency make up per Attachment A. Emergency sources listed in Attachment A are RWST, CST or Fire Protection.

A is incorrect but plausible as this source could be used if not an emergency.

C is incorrect but plausible as this source could be used if not an emergency.

D is incorrect but plausible as this source could be used if not an emergency.					
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1215.07, Loss of Spent Fuel Pool Cooling (Rev 13)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L1192I08RO				(As available)
Question Source:	Bank #	X	23150		
	Modified Bank#				(Note changes or attach Parent)
	New				
Question History:	Last NRC Exam	2005 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	(CFR: 41.5)			
	55.43				
Comments:					

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Examination Outline Cross-reference:	Level	RO	SRO
Q58	Tier #	2	
	Group #	2	
	K/A #	**034** Fuel Handling Equipment System (FHES) **A4** Ability to manually operate and/or monitor in the control room: **A4.02** Neutron levels	
	Importance Rating	3.5	

Proposed Question:

During refueling operations, the MCB RO notices that the Source Range Audio Count Rate Monitor in the control room is no longer functioning.

What actions should be taken?

A. Immediately suspend all operations involving core alterations or positive reactivity changes.

B. Refueling operations may continue if an operable audio count rate monitor is available in containment.

C. Immediately suspend core alterations and evacuate containment until the audio count rate monitor is returned to operable status.

D. Ensure the boron concentration in the RCS is greater than the Refueling Boron Concentration Limits of the COLR once per 12 hours and suspend any activities which could dilute the RCS.

Proposed Answer: A

Explanation (Optional):

A is correct. Per T.S.3.9.2 continuous audible indication is required in the containment and the control room. If not immediately suspend core alterations or positive reactivity changes.

B is incorrect but it is plausible that with audible count rate indicating in the containment then the SR is operable and refueling operations could continue.

C is incorrect but it is plausible to evacuate containment as the SR instruments sound the containment evacuation alarm on high count rate.

D is incorrect but plausible as this is part of the required actions of T.S.3.9.2 action b if both SR instruments are inoperable.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.9.2 Refueling Operations Instrumentation (Amendment No. 93)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8030I12RO			(As available)
Question Source:	Bank #	X	TEB16366	
	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	(CFR: 41.7)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q59	Tier #	2	
	Group #	2	
	K/A #	**035** Steam Generator System (S/GS) **K6** Knowledge of the effect of a loss or malfunction on the following will have on the S/GS: **K6.01** MSIVs	
	Importance Rating	3.2	

Proposed Question:

The following plant conditions exist:

- Plant startup is in progress.
- The reactor is critical at the point of adding heat.
- Tavg is 557°F.
- Three MSIVs are fully open, but the fourth cannot be opened by any means.

What restrictions should be placed on the startup, if any? (Reference provided)

A. None.
B. The startup can continue but power is restricted to <5%.
C. The startup can continue but power is restricted to 10%.
D. The plant must be cooled down to < 350°F within 6 hours.

Proposed Answer: B

Explanation (Optional):

B is correct. T.S.3.7.1.5 Action for Mode 2 and 3 states with one MSIV inoperable Subsequent mode 2 or 3 operation may continue provided the valve is maintained closed. Continuation of startup to mode one (> 5%) is not allowed as mode one action requires valve to be open if inoperable and operability restored within 4 hours.

A is incorrect but plausible as with current conditions (valve closed) actions for mode 2 are complete and operation with one valve inoperable is allowed in mode 1.

C is incorrect but it is plausible that operation up to 10% could be maintain as this is the power that

operation on the ASDVs could support if all MSIVs were required to be closed. D is incorrect but plausible if the conditions above were not allowed. Actions for mode 1,2 and 3 require entry to HOT STANDBY with in 6 hours.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.7.1.5, Plant Systems, Turbine Cycle, Main Steam Isolation Valves (Rev None)			
Proposed references to be provided to applicants during examination:				T.S.3.7.1.5
Learning Objective:	L8041I13RO			(As available)
Question Source:	Bank #	X	TEB6027	
	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: 41.7)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q60	Tier #	2	
	Group #	2	
	K/A #	**041** Steam Dump System (SDS) and Turbine Bypass Control **A3** Ability to monitor automatic operation of the SDS, including: **A3.02** RCS pressure, RCS temperature, and reactor power	
	Importance Rating	3.3	

Proposed Question:

The following plant conditions exist:

- The plant is operating at 12% power, Middle of Life (MOL).
- Control Bank D rods are at 150 steps withdrawn in MANUAL.
- Main turbine is steady at 800 RPM during turbine startup.
- Steam Dumps are in the STEAM PRESSURE Mode of control and set for 1050 psig in AUTO.
- MS-PK-507, main steam dump controller AUTO setpoint is lowered to 1040 psig.

What effect will this have on Tavg and reactor power assuming NO other operator action?

A. Tavg will RISE, Reactor Power will RISE.
B. Tavg will LOWER, Reactor Power will RISE.
C. Tavg will RISE, Reactor Power will LOWER.
D. Tavg will LOWER, Reactor Power will LOWER.

Proposed Answer: B

Explanation (Optional):

B is correct. With Steam Dumps operating in Automatic in the Steam Pressure mode the controller compares steam pressure to setpoint. If not at setpoint the controller will take action to open or close steam dumps to bring steam pressure to setpoint. Lowering setpoint to 1040 psig will reduce setpoint below actual steam pressure and the controller output will increase to open steam dump valves to lower actual steam pressure. Steam flow to the main condenser will increase. Tavg will decrease. The reduced temperature will be a positive reactivity change and reactor power will

increase.

A is incorrect but is plausible as reactor power will increase and program Tavg increases with increasing reactor power.

C is incorrect but plausible that the controller could attempt to lower steam pressure by reducing controller output.

D is incorrect but plausible as Tavg will lower and that the controller could attempt to lower steam pressure by reducing controller output.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Steam Dump Detailed Systems Text Fig 4.3 and 4.4 (Rev 6/05)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8047I06RO			(As available)
Question Source:	Bank #	X	TEB30810	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Seabrook 2005		

(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: 41.7)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q61	Tier #	2	
	Group #	2	
	K/A #	**045** Main Turbine Generator (MT/G) System **2.4.49** Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.	
	Importance Rating	4.6	

Proposed Question:

During E-0, "Reactor Trip or Safety Injection" immediate actions, the Main Turbine fails to trip automatically or manually.

What is the next required action?

A. Manually open the generator breaker.
B. Dispatch an NSO to locally trip the turbine.
C. Close the MSIVs. When generator output is ZERO MWe then manually open the generator breaker.
D. Manually run back the turbine, when generator output is ZERO MWe then manually open the generator breaker.

Proposed Answer: C

Explanation (Optional):

C is correct. E-0, Reactor Trip or Safety Injection, step 2a, RNO directs the operator to manually trip the turbine. If the turbine will not manually trip the step then directs the operator to "close the MSIV's. When generator output is ZERO Mwe, then open the generator breaker".

A is incorrect but plausible as it is part of the procedure step, however, it is not opened until after the turbine is tripped or MSIVs closed and generator output is checked to be zero.

B is incorrect but plausible as this is an action in FR-S.1 step 6 when checking if the turbine trip has occurred.

D is incorrect but plausible as manually running back the turbine is a strategy in the FR-S.1 background document to reduce steam flow.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-0, Reactor Trip or Safety Injection Background Information for WOG ERG FR-S.1 Response to Nuclear Power Generation/ATWS (HP-Rev 2)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1202I04RO			(As available)
Question Source:	Bank #	X	TEB29943	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2007 NRC		
(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	(CFR: 41.10)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q62	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:

Plant conditions:

- 75% power.
- The UAT feeder breaker to Bus 3 opened <u>and</u> the RAT feeder breaker failed to close.
- 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.
- D7761, CTL ROD BANK D INSERTION LIMIT LOW is in alarm.
- D4421, TAVG - TREF DEVIATION, is in alarm.

What procedure should the crew enter?

A. OS1202.04, "Rapid Boration".
B. OS1231.04, "Rapid Down Power".
C. OS1231.03, "Turbine Runback/Setback".
D. OS1290.02, "Response to Secondary System Transient".

Proposed Answer:	C	

Explanation (Optional):

C is correct. With the loss of Bus 3 two condensate pumps (CO-P-30A and C) are lost. At 75% power the loss of two condensate pumps meets the logic for turbine setback and entry conditions for OS1202.04.

A is incorrect but plausible as Rods inserting in auto during a runback may cause the rods to insert below the Rod insertion limit. Rod Insertion Limit LOW-LOW alarm would require Rapid Boration. Question stem only has the RIL LOW alarm in.

B is incorrect but plausible as Rapid Down power of plant due to loss of all but a single condensate could avert a MFP trip on low suction pressure.

D is incorrect but plausible because if non-vital bus 4 were lost then only one condensate pump would have been lost and the standby pump would start. This is an entry condition for OS1290.02.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1231.03, Turbine Runback/Setback (Rev 20)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1183I09RO			(As available)
Question Source:	Bank #	X	TEB23090	
	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: 41.5)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q63	Tier #	2	
	Group #	2	
	K/A #	**068** Liquid Radwaste System (LRS) **K1** Knowledge of the physical connections and/or cause effect relationships between the Liquid Radwaste System and the following systems: **K1.07** Sources of liquid wastes for LRS	
	Importance Rating	2.7	

Proposed Question:

Which ONE (1) of the following describes three sources of liquid waste to the Reactor Coolant Drain Tank?

A. Containment trench.
   Reactor vessel head seal ring
   RCP #2 seal.

B. "A" Containment Sump.
   RCP #1 seal.
   Excess letdown heat exchanger divert.

C. Reactor vessel head seal ring.
   RCP #2 seal.
   Excess letdown heat exchanger divert.

D. RCS Loop drains.
   "A" PORV block valve (RC-V-122), valve stem leak off.
   CAH-F-1A cooler leak-off.

Proposed Answer: C

Explanation (Optional):

C is correct. Reactor vessel seal ring, RCP#2 seal and Excess letdown HX divert flow go to the

RCDT.

A is incorrect but plausible. Reactor vessel seal ring and RCP #2 seals do got to RCDT and the RCDT can be drained to the containment trench not the trench to the RCDT.

B is incorrect but plausible. Excess letdown divert does go to RCDT. RCP #1 seal can return to PRT if containment flow path isolated via seal return relief. RCDT relief is direct to the "A" containment sump.

D is incorrect but plausible. RCS loop drains can be directed to RCDT pump suctions, PORV packing does go to RCDT but CAH-F-1A coolers drain to trench.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	PID-1-WLD-B20218, Waste Processing Liquid Drains Reactor Coolant Systems (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	(CFR: 41.3)	
	55.43		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q64	Tier #	2	
	Group #	2	
	K/A #	**075** Circulating Water System **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.01** Loss of intake structure	
	Importance Rating	3.0*	

Proposed Question:

Plant conditions:

- 100% power.
- Following a record setting Nor'easter the following conditions are observed:
  - Increased vibrations observed locally on CW-P39A, B and C.
  - Frequent CW Lube water low flow alarms.
  - Cyclone separator drain lines full of sand.

Subsequently lube water is lost to <u>all</u> 3 CW pumps and cannot be restored.

Which of the following describes the actions to respond to this event?

A. Place the control switches for all 3 CW pumps in stop. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". When immediate actions are complete close the MSIVs.

B. Place the control switches for all 3 CW pumps in stop. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". When immediate actions are complete place either Steam Dump Interlock selector switch in OFF.

C. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". When immediate actions are complete place the control switches for all 3 CW pumps in stop. Close the MSIVs.

D. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". When immediate actions are complete place the control switches for all 3 CW pumps in stop. Place either Steam Dump Interlock selector switch in OFF.

Proposed Answer:	C	

Explanation (Optional):

KA match: Sand/silt intrusion that requires stopping all CW pumps is equivalent to a loss of intake structure.

C is correct. Loss of lube water flow to all CW pumps requires securing all CW pumps. ON1238.01 requires reactor to be tripped and E-0 immediate actions completed prior to securing all CW pumps and then MSIVs to be closed to protect condenser from overpressure with no CW.

A is incorrect but plausible. Reactor must be tripped prior to securing CW pumps.

B is incorrect but plausible. Reactor must be tripped prior to securing CW pumps. Placing steam dumps in off could provide another method to protect condenser from overpressure with no CW. However procedure requires closing MSIVs.

D is incorrect but plausible. Placing steam dumps in off could provide another method to protect condenser from overpressure with no CW. However procedure requires closing MSIVs.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1238.01 Circulating Water Malfunction (Rev 15)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1188I14RO			(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	(CFR: 41.5)	
	55.43		

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
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Examination Outline Cross-reference:	Level	RO	SRO
Q65	Tier #	2	
	Group #	2	
	K/A #	**086** Fire Protection System (FPS) **A4** Ability to manually operate and/or monitor in the control room: **A4.05** Deluge valves	
	Importance Rating	3.0	

Proposed Question:

The deluge valve for the "A" UAT spuriously actuates the fire suppression system for the "A" UAT.

What indications are available to the control room operators of this condition and how is fire suppression secured?

A. Deluge Flow alarm with Electric Fire Pump start ONLY. Close manual Isolation and manually reset deluge valve.

B. Deluge Flow alarm with Diesel Fire Pump start ONLY. Close manual Isolation and manually reset deluge valve.

C. Deluge Flow alarm with Electric Fire Pump start ONLY. Restore control air to deluge valve actuator and observe valve closure.

D. Deluge Flow alarm with Diesel Fire Pump start ONLY. Restore control air to deluge valve actuator and observe valve closure.

Proposed Answer: A

Explanation (Optional):

A is correct. If the UAT deluge system actuates a flow alarm and the Electric Fire pump would start. Electric and Diesel pumps are in a standby condition. The Electric pump has a shorter time delay (2 sec) on low pressure automatic start than the two Diesel driven pumps (10 sec and 20sec). Securing suppression requires manual actions to isolate and reset the deluge valve.

B is incorrect but it is plausible that on deluge system actuation that a diesel driven fire pump would start. But if the diesel fire pump starts then the electric fire pump should have started also.

C is incorrect but it is plausible since pre-action sprinkler systems have supervisory air or nitrogen pressure on the system and it could be confused as control air required for transformer yard deluge system actuator control air.

D is incorrect but it is plausible since pre-action sprinkler systems have supervisory air or nitrogen pressure on the system and it could be confused as control air required for transformer yard deluge system actuator control air.					
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	PID-1-FP-B20269, Fire Protection Turbine Building Detail Sh 1 of 2 (Rev 15) ON0443.40, Transformer Yard Deluge Systems 18 Month Operability Tests (Rev 08) OS0043.08, Resetting of Grinnel Multimatic Valves Model A-4 (Rev 04 Chg 04) OX0443.13, Fire Pumps 18 Month Auto Start Test (Rev 1)				
Proposed references to be provided to applicants during examination:				None	
Learning Objective:	L8089I05RO			(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	(CFR: 41.7)			
	55.43				
Comments:					

Seabrook Station 2013 Licensed Operator NRC Written Exam
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Examination Outline Cross-reference:	Level	RO	SRO
Q66	Tier #	3	
	Group #		
	K/A #	**2.1.5** Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.	
	Importance Rating	2.9*	

Proposed Question:

If a licensed control room operator is designated as No Solo Operation, this means he or she:

A. Cannot be allowed to perform any complex evolutions by themselves.

B. Must have some other licensed individual present in the control room while on watch.

C. Must have an SRO licensed supervisor present during all operations that they are involved in.

D. Cannot be allowed to manipulate <u>any</u> controls affecting reactivity or plant power level without an SRO licensed individual present.

Proposed Answer: B

Explanation (Optional):

B is correct. OPMM chapter 1 section 5.3 defines the no solo operation policy. Another licensed individual must be present when the operator is performing licensed activities. This individual's responsibility is to call for help and assistance if required.

A is incorrect but it is plausible that only complex evolutions require and additional operator.

C is incorrect but plausible the additional operator must be a supervisor.

D is incorrect but plausible additional operator required for reactivity or power changes.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Operations Management Manual CH1 Sect 5 Policies (Rev 86)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1505I06RO			(As available)
Question Source:	Bank #	X	TEB6633	
	Modified Bank#			(Note changes or attach Parent)

	New			
Question History:	Last NRC Exam	2009 Seabrook NRC		
(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	(CFR: 41.10)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q67	Tier #	3	
	Group #		
	K/A #	**2.1.6** Ability to manage the control room crew during plant transients.	
	Importance Rating	3.8*	

Proposed Question:

While performing OS1231.04, "Rapid Down Power", what guidance is provided for human performance protocols?

A. The US must direct and monitor control rod manipulations and borations. Peer checks may be waived during transients.

B. The US must direct and monitor control rod manipulations and borations. Peer checks are required for control rod manipulations and borations.

C. The US may direct the operators to perform reactivity changes rather than direct each discrete reactivity manipulation. Peer checks may be waived during transients.

D. The US may direct the operators to perform reactivity changes rather than direct each discrete reactivity manipulation. Peer checks are required for control rod manipulations and borations.

Proposed Answer: C

Explanation (Optional):

C is correct. OS1231.04 Rapid Down Power brief allows the US to direct operators to perform reactivity changes rather than direct discrete reactivity manipulations and to waive peer checks.

A is incorrect but plausible as peer checks may be waived and during normal operations the US must monitor reactivity changes.

B is incorrect but plausible as peer checks are required and reactivity changes must be directly monitored during normal operations.

D is incorrect but plausible as operators performing reactivity changes vice being directed for discrete manipulations is allowed for rapid down powers and peer checks are required during normal operations.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1231.04, Rapid Down Power (Rev 06)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1197I08RO			(As available)
Question Source:	Bank #	X	TEB30127	
	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	(CFR: 41.10)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q68	Tier #	3	
	Group #		
	K/A #	**2.2.20** Knowledge of the process for managing troubleshooting activities.	
	Importance Rating	2.6	

Proposed Question:

Per MA-AA-100-1011, Equipment Troubleshooting, which TWO (2) of the following are classified as intrusive?

1. Clamp on ammeter readings
2. Lifting leads for voltmeter readings
3. Vibration diagnostic equipment readings
4. Temporary pressure gage connected to plant system

A. 1 and 3
B. 2 and 4
C. 2 and 3
D. 1 and 4

Proposed Answer: B

Explanation (Optional):

B is correct. Per MA-AA-100-1011 lifting leads and connecting temporary test gages are intrusive activities.

A is incorrect but is plausible as connecting temporary test gages is intrusive and clamp on ammeters could be considered intrusive.

C is incorrect but is plausible as lifting leads is intrusive and connecting temporary test gages could be non intrusive as they are connected to existing test valves.

D is incorrect but is plausible as connecting gages is intrusive. And lifting leads for voltage readings could be non intrusive. A recent change to the trouble shooting procedure changed connecting voltmeters to a nonintrusive activity but lifting leads for voltmeter readings is still

intrusive.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	MA-AA-100-1011, Equipment Troubleshooting (Rev 0)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1514I21RO			(As available)
Question Source:	Bank #	X	TEB30545	
	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	(CFR: 41.10)		
	55.43			
Comments:				

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Examination Outline Cross-reference:	Level	RO	SRO
Q69	Tier #	3	
	Group #		
	K/A #	**2.2.22** Knowledge of limiting conditions for operations and safety limits.	
	Importance Rating	4.0	

Proposed Question:

Which of the following describes ONLY components assumed to operate at their setpoints to prevent exceeding the Technical Specification safety limit on RCS pressure?

A. Pressurizer PORVs and ASDVs.

B. Pressurizer Safety Valves and ASDVs.

C. Pressurizer PORVs and Main Steam Safety Valves.

D. Pressurizer Safety Valves and Main Steam Safety Valves.

Proposed Answer: D

Explanation (Optional):

K/A match: K/A 2.2.22 has limiting condition AND safety limits. Unable to include LCO in question as there are no LCOs applicable to safety limits.

D is correct. T.S. SL 2.1.2 Bases. The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

More specifically, no credit is taken for operation of any of the following:
a. Pressurizer power operated relief valves (PORVs),
b. Steam line relief valve,
c. Steam Dump System,
d. Reactor Control System,
e. Pressurizer Level Control System, or
f. Pressurizer spray valve.

A is incorrect but plausible. PORVS and ASDVs mitigate RCS pressure transients and have T.S. requirements but are not credited for operation in the safety analysis.

B is incorrect but plausible. ASDVs mitigate RCS pressure transients and have T.S. requirements but are not credited for operation in the safety analysis.

C is incorrect but plausible. PORVs mitigate RCS pressure transients and have T.S. requirements but are not credited for operation in the safety analysis.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S. Basis 2.1.2 (Rev BC 08-02)			
Proposed references to be provided to applicants during examination:				
Learning Objective:	L8010I05RO			(As available)
Question Source:	Bank #	X	Robinson	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2003		
(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	(CFR: 41.5)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
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Examination Outline Cross-reference:	Level	RO	SRO
Q70	Tier #	3	
	Group #		
	K/A #	**2.2.38** Knowledge of conditions and limitations in the facility license.	
	Importance Rating	3.6	

Proposed Question:

Emergency makeup to the spent fuel pool from the CST is permissible because:

A. Restrictions on placement of fuel assemblies ensure that the SFP will remain <.95 keff when flooded with unborated water.

B. Boraflex between spent fuel assemblies prevents criticality when flooded with unborated water regardless of fuel assembly placement.

C. Restrictions on fuel assembly burnup ensure there are insufficient source neutrons available to cause criticality when flooded with unborated water.

D. Restrictions on decay time of fuel assemblies ensure there are insufficient source neutrons available to cause criticality when flooded with unborated water.

Proposed Answer: A

Explanation (Optional):

A is correct. T.S.3.9.13 figure 3.9-1 applies limits on fuel assembly enrichment and burnup with assembly placement. These restrictions ensure the Keff of the Spent Fuel Pool remains <.95 when flooded with unborated water.

B is incorrect but plausible as boraflex is applied between storage rack cells. However administrative controls on placement of fuel assemblies are required in the analysis.

C is incorrect but plausible. There is restrictions on assembly burnup and source neutrons would be required for subcritical multiplication to increase Keff >.95. However source strength is not what is controlled but the enrichment of fuel assemblies and their locations.

D is incorrect but plausible. There is an 80 hour delay time for movement of fuel and source neutrons would be required for subcritical multiplication to increase Keff >.95. However source strength is not what is controlled but the enrichment of fuel assemblies and their locations.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.9.13, Refueling Operation, Spent Fuel Assembly Storage (Amendment No.6) Basis T.S.3.9.13, Spent Fuel Assembly Storage (Rev 08-10) UFSAR 9.1.2, Auxiliary Systems, Fuel Storage and Handling

	(Rev 15)				
Proposed references to be provided to applicants during examination:					None
Learning Objective:	L8060I11RO				(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	(CFR: 41.7)			
	55.43				
Comments:					

Examination Outline Cross-reference:	Level	RO	SRO
Q71	Tier #	3	
	Group #		
	K/A #	**2.3.4** Knowledge of radiation exposure limits under normal or emergency conditions.	
	Importance Rating	3.2	

Proposed Question:

Given the following conditions:

- A General Emergency has been declared due to a LOCA Outside Containment.
- An operator volunteers to make an emergency entry into the penetration area to isolate the leak. Isolating the leak would result in a significant reduction in offsite dose, protecting "large populations".
- The operator has been briefed and is fully aware of the risks involved.

What is the maximum exposure that may be authorized for this situation?

A. 4000 mrem TEDE

B. 5000 mrem TEDE

C. 10000 mrem TEDE

D. May exceed 25000 mrem TEDE

Proposed Answer: D

Explanation (Optional):

D is correct. Per ER4.3, Figure 2, Emergency Dose Limits an individual may be authorized to exceed 25000 mrem TEDE for lifesaving or protection of large populations "only on a voluntary basis to persons fully aware of the risks involved".

A is incorrect but plausible. 4000 mrem TEDE is the maximum dose allowed prior to needing the Plant Manager's approval for an exposure limit upgrade to 4000-5000 mrem under normal plant conditions, per RP5.1, Figure 5.3, Exposure Limit Upgrades for All Personnel.

B is incorrect but plausible. This is the normal federal limit for TEDE. Additionally, it is listed in ER4.3, Figure 2, Emergency Dose Limits, as the limit for "all activities".

C is incorrect but plausible. 10000R is the previous limit for protecting plant equipment.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ER 4.3, Radiation Protection During Emergency Conditions (Rev 30) RP 5.1, Annual Occupational Exposure Control and Increases Radiation Exposure Approval (Rev 20)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1525I13RO			(As available)
Question Source:	Bank #	X	TEB31028	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 NRC Remediation 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	(CFR: 41.12)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q72	Tier #	3	
	Group #		
	K/A #	**2.3.12** Knowledge of radiological safety principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.	
	Importance Rating	3.2	

Proposed Question:

Plant conditions:

- Refueling outage in progress.
- Incore Instrument thimbles have been withdrawn.
- Core off-load has commenced.
- The crew has noted a one-inch Refueling Cavity inventory loss during the first eight hours of the shift.
- "B" Containment sump run times are more frequent than expected.
- The crew is determining if under-vessel inspection is possible.

What is the concern, and who can authorize under-vessel access, if at all?

A. The highly irradiated Incore Instrument thimbles are withdrawn into the area under the reactor vessel. Access cannot be authorized by anyone.

B. During fuel transfer, under-vessel access is restricted due to rapidly changing dose rates. RP Manager must specify conditions and authorize access.

C. The highly irradiated Movable Fission Chambers are stored in the instrumentation tunnel. Plant General Manager <u>and</u> RP Manager must both authorize access.

D. A detached, highly irradiated Movable Fission Chamber is present in thimble #36. It can move freely in its guide tube. RP Manager must specify conditions and authorize access.

Proposed Answer: A

Explanation (Optional):

A is correct. RP 9.2 Fig 5.1 specifies NO ENTRY ALLOWED beneath the Reactor Vessel with incore thimbles withdrawn. Irradiated incore thimbles cause very high radiation fields and will

normally be posted "Grave Danger: Very High Radiation Area"

B is incorrect but is plausible as movement of fuel assemblies cause rapidly changing radiation levels in the vicinity of the fuel transfer tube and could also be assumed in areas under the reactor vessel as well. RP manager authorization is required for exposure limit upgrades.

C is incorrect but plausible as there are Movable Fission Chambers stored in their storage location in containment. It is possible the storage location could be the instrument tunnel under the vessel. RP manager and PGM authorization is required for exposure limit upgrades.

D is incorrect but plausible as there is a detached movable fission chamber in thimble #36. It is able to move freely in its guide tube. RP does survey each outage below vessel to assertian its location. RP manager authorization is required for exposure limit upgrades.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	RP 9.2, Radiological Access Requirements to Containment Areas (Rev 13)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8028I05RO, L1525I09RO			(As available)
Question Source:	Bank #	X	TEB31453	
	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	(CFR: 41.12)	
	55.43		

Comments:

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Examination Outline Cross-reference:	Level	RO	SRO
Q73	Tier #	3	
	Group #		
	K/A #	**2.4.8** Knowledge of how abnormal operating procedures are used in conjunction with EOPs.	
	Importance Rating	3.8	

Proposed Question:

Given the following:

- The reactor trips from 100% power.
- Concurrent with the reactor trip controlling PZR level instrument L-459 fails low and letdown isolates.

How is the crew required to implement the EOPs/AOPs?

A. The crew enters E-0, "Reactor Trip or Safety Injection". After completing step 4, the crew transitions to ES-0.1, "Reactor Trip Response" and completes OS1201.07, "PZR Level Instrument Failure" in parallel with ES-0.1.

B. The US hands the WCS OS1201.07, "PZR Level Instrument Failure", who directs the BOP to perform the steps of the AOP in parallel with the US entering E-0, "Reactor Trip or Safety Injection" and directs the PSO to perform the steps of E-0.

C. The crew enters E-0, "Reactor Trip or Safety Injection". After completing step 4, the crew transitions to OS1201.07, "PZR Level Instrument Failure", when AOP is completed the crew transitions to ES-0.1, "Reactor Trip Response".

D. The US directs the PSO to take manual control of charging and letdown and maintain PZR level on program. The crew then enters E-0, "Reactor Trip or Safety Injection". After completing step 4, the crew transitions to ES-0.1, "Reactor Trip Response" and completes OS1201.07, "PZR Level Instrument Failure" in parallel with ES-0.1.

Proposed Answer: A

Explanation (Optional):

A is correct. OP 9.2 section 4.9.4 states there are times when the use of other procedures will provide written guidance to respond to a condition not addressed in the EOP network. Parallel use of abnormal procedures is allowed following completion of immediate actions.

B is incorrect. It is plausible that as immediate action steps are being completed by the PSO and the US an additional crew member and the BOP could perform abnormal procedure actions.

C is incorrect. It is plausible to transition to the abnormal procedure as this is done if not in the Emergency procedures.

D is incorrect but plausible take simple and quick skill of the operator actions to control pressurizer level by reducing charging prior to entering E-0.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OP 9.2, Transient Response Procedure Guide (Rev 16)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1505I10RO			(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	(CFR: 41.10)		
	55.43			
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q74	Tier #	3	
	Group #		
	K/A #	**2.4.41** Knowledge of the emergency action level thresholds and classifications.	
	Importance Rating	2.9	

Proposed Question:

While operating at 100% the following sequence of events occurred:

T=1	An RCS leak to Containment occurred.
T=2	The leakage flow rate was verified to be 30 gpm, but the source unknown.
	The crew entered the appropriate AOP.
T=3	The source of the leak could not be isolated.
	The SM ordered a plant shutdown.
T=4	During the shutdown, the leak was determined to be pressure boundary leakage.

The SM should have declared the Emergency Classification within 15 minutes of when the leak _____.

Which of the following correctly completes the above statement? (Reference material provided)

A. first occurred (T=1).

B. flow rate was verified (T=2).

C. was determined to be <u>not</u> isolable (T=3).

D. was determined to be pressure boundary leakage (T=4).

Proposed Answer: B

Explanation (Optional):

B is correct. Unidentified/pressure boundary leakage > 10gpm OR identified leakage > 25gpm meet requirements for UE declaration (SU5). 15 minutes is the required time to declare the event from initial leak rate determination of 30 gpm from an unknown source.

A is incorrect but it is plausible to think time is from when the leak first occurred. ER-1.1 states the time is from when the available indications to the plant operators that an EAL has been exceeded.

C is incorrect but plausible. If leakage is isolable from the RCS then it is no longer considered RCS leakage and the EAL does not apply. For this question it was determined not to be isolable and the declaration time is from the initial leak rate determination.

D is incorrect but plausible. Pressure boundary leakage is part of the EAL. However it was not determined to be so until after shutdown started. Declaration is required from initial leak rate determination from unknown source.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ER 1.1 Classification of Emergencies. (Rev 52)

Proposed references to be provided to applicants during examination:				ER-1.1
Learning Objective:	L1509I22RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	TEB31100	(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: 41.10)	

	55.43	

Comments:
TEB31100 prior to modification:

Plant conditions:

- While operating at 100% power, an RCS leak to Containment occurred.
- The leakage flow rate was immediately verified to be 30 gpm, but the source unknown.
- The crew entered the appropriate AOP.
- The source of the leak could not be isolated.
- The SM ordered a plant shutdown.
- During the shutdown, the leak was determined to be pressure boundary leakage

Which, if any, Emergency Classification must be declared and when? (Reference material provided)

A No classification is required.

B UE, within 15 minutes of when the leak flow rate was verified.

C UE, within 15 minutes of when the leak was determined to be <u>not</u> isolable.

D UE, within 15 minutes of when the leak was determined to be pressure boundary leakage.

Answer: B

Examination Outline Cross-reference:	Level	RO	SRO
Q75	Tier #	3	
	Group #		
	K/A #	**2.4.50** Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	
	Importance Rating	4.2	

Proposed Question:

Plant conditions:

- Reactor trip from 100% power.
- ES-0.1, "Reactor Trip Response" is being implemented.
- While throttling EFW to "A" SG, EFW to "B" and "C" SGs automatically isolated.
- The following VAS alarms are received:
  - F5281, SG B EFW FLOW HIGH.
  - F5449, SG C EFW FLOW HIGH.
  - F5453, SG D EFW FLOW HIGH.
- The following valves are closed:
  - 1-FW-FV-4224-A, SG "B" EFW Isolation valve (Train A).
  - 1-FW-FV-4234-B, SG "C" EFW Isolation valve (Train B).
- EFW flow to "A" and "D" SGs has been throttled to 400 gpm each.

What actions are necessary to restore EFW to all SGs?

A. Momentarily place the non-isolated SG B and C Train "A" <u>and</u> "B" EFW valve switches to THROTTLE OPEN. Restore flow to "B" and "C" SGs as required.

B. Momentarily place the non-isolated SG B and C Train "A" <u>and</u> "B" EFW valve switches to THROTTLE CLOSE. Restore flow to "B" and "C" SGs as required.

C. Momentarily place the non-isolated SG B, C, and D Train "A" <u>and</u> "B" EFW valve switches to THROTTLE OPEN. Restore flow to "B" and "C" SGs as required.

D. Momentarily place the non-isolated SG B, C, and D Train "A" <u>and</u> "B" EFW valve switches to THROTTLE CLOSE. Restore flow to "B" and "C" SGs as required.

Proposed Answer:	C

Explanation (Optional):

When an EFW high flow isolation occurs (565 gpm) on a SG the other three SG isolation valves in the same train are blocked from closing. This prevents a cascading isolation of all EFW. To reset the isolation, the control switches must be placed in the throttle open position. If the isolation signal is not reset for an already open valve that is blocked from closing it will close when the isolated valve is taken to open.

C is correct. Per VPRO guidance once EFW flow is < 510 gpm then the Non-isolated flow control switches are momentarily placed in the throttle OPEN position to reset the locked in isolation signal. Then EFW flow is re-established. If non-isolated valves for SG B, C and D are not placed in throttle OPEN the trip close signal does not get reset and the open B, C and D SG EFW isolation valves will close.

A is incorrect but plausible that only the valves in the affected SGs need to be taken to OPEN to reset all the isolation signals. D SG isolation valves have an isolation signal in but are blocked from closing due to the already closed valves in B and C SGs.

B is incorrect but plausible that only the isolation valves in the affected SGs are required to be placed momentarily in CLOSE to reset the isolation signal.

D is incorrect but plausible tht the isolation reset is accomplished by taking control switches to the throttle CLOSE position.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	VPRO F5453, SG D EFW Flow High (Rev 5) VPRO F5449, SG C EFW Flow High (Rev 5) VPRO F5281, SG B EFW Flow High (Rev5)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8045I06RO			(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	(CFR: 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:

Plant conditions:

- Reactor has tripped from 100% power.
- ES-0.1, "Reactor Trip Response" has been completed.
- OS1000.11, "Post Trip to Hot Standby" is being performed.
- NOP and NOT.
- CST level has decreased to 290,000 gallons.
- NO makeup source is available to the CST.
- SUFP has been placed in service.

What actions, if any, are required to meet UFSAR assumptions?

A. Timely performance of plant cooldown is required to have CST level available and SG NR levels at 50% when RHR is placed in service.

B. No action required. CST level of 200,000 gallons assures sufficient inventory to reach RHR under all conditions.

C. No action required. CST level of 200,000 gallons is sufficient inventory to remain in Hot standby.

D. CST must be shifted to the lower tap with the low suction pressure trip bypassed.

Proposed Answer: A

Explanation (Optional):

A is correct. Precaution 3.4 of OS1000.11 states if CST make up is unavailable and level is <290,000 gallons timely performance of plant cooldown will ensure CST available until RHR is placed in service. VPRO D4120 CST Low Level states 290,000 gallons allows 4 hours in hot

standby, 4 hours to cooldown to RHR, 1 hour to place RHR in service and still have 50% NR SG level.

B is incorrect but plausible as UFSAR minimum volume required is 194,000 gallons. However 17,648 gallons in the CST is unusable due to draft of floating cover, inventory trapped, vortexing and instrument inaccuracies.

C is incorrect but it is plausible you can remain in hot standby.

D is correct but plausible as this is an action in ES-0.1 if required to use SUFP as a source of EFW and CST level is <250,000 gallons.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1000.11, Post Trip to Hot Standby (Rev 11) OS1000.04, Plant Cooldown From Hot Standby to Cold Shutdown (Rev 40) ES0.1, Reactor Trip Response (Rev 37) VPRO D4120, Condensate Storage Tank Level Low (Rev 06) UFSAR 9.2.6, Auxiliary Systems, Water Systems, Condensate Storage Facility (Rev 15)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1169I03RO			(As available)
Question Source:	Bank #	X	TEB24485	
	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	(CFR: 43.2)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

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: **EA2.15** RCS parameters	
	Importance Rating		3.4

Proposed Question:

Plant conditions:

- A Small Break LOCA has occurred.
- The crew is performing the actions of ES-1.2 "Post LOCA Cooldown and Depressurization".
- ECCS pumps have been stopped.
- Normal Charging is aligned.
- The crew is depressurizing the RCS.

When the depressurization is stopped, the following conditions exist:

- RCS Subcooling is 37°F and decreasing.
- Pressurizer Level is 18% and decreasing.

What actions should be taken?

A. Manually start ECCS pumps as necessary to regain subcooling.
B. Increase RCS pressure using pressurizer heaters to regain subcooling.
C. Isolate Letdown. Check to ensure Pressurizer Level stabilizes above 5%.
D. Actuate Safety Injection and verify all safeguards equipment has actuated.

Proposed Answer: A

Explanation (Optional):

A is correct. ES-1.2, "Post LOCA Cooldown and Depressurization", step 22 and OAS page, Item 1, ECCS Reinitiation Criteria both direct manually aligning valves and starting ECCS pumps as necessary if RCS subcooling is less than 40°F or Pressurizer level is less than 7%.

B is incorrect but plausible. It is true that pressurizer heaters are operated during the depressurization process however this is done to establish saturated conditions in the pressurizer to maintain a steam bubble. The heaters are not used as part of the strategy to recover subcooling.

C is incorrect but plausible. The procedure does direct controlling charging flow as necessary to maintain Pressurizer level during the depressurization process. It is conceivable that charging and letdown could be utilized as part of the pressurizer level control strategy however letdown flow is not established in ES-1.2.

D is incorrect but plausible. It is true that the strategy to recover subcooling includes restarting ECCS equipment however a complete reinitiation of SI is not directed as this would result in a higher RCS pressure than necessary for the given plant conditions.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-1.2 Post LOCA Cooldown and Depressurization (Rev 38)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1204I06RO			(As available)
Question Source:	Bank #	X	TEB23182	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	Seabrook 2010		
(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	(CFR: 43.5)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q78	Tier #		1
	Group #		1
	K/A #	**038** Steam Generator Tube Rupture (SGTR) **EA2** Ability to determine or interpret the following as they apply to a SGTR: **EA2.07** Plant conditions, from survey of control room indications	
	Importance Rating		4.8

Proposed Question:

Plant conditions:

- SGTR on the "D" SG.
- The crew entered E-3, "Steam Generator Tube Rupture".
- RCS cooldown and depressurization are completed.
- The crew is checking to determine if ECCS flow should be terminated.
- Current conditions are as follows:
  - RCS subcooling is 45°F.
  - RCS pressure is slowly decreasing.
  - PZR level is 20% and slowly decreasing.
  - "D" SG pressure is 1100 psig and stable.
  - SG NR levels are as follows:
    - "A" SG 5% and increasing
    - "B" SG 4% and increasing
    - "C" SG 6% and increasing
    - "D" SG 55% and decreasing

What action is the crew required to perform?

A. Stop ECCS pumps. Remain in E-3 and establish normal charging.

B. Do not stop ECCS pumps. Go to ECA-3.1, "SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired".

C. Do not stop ECCS pumps. Transition to FR-H.1, "Response to Loss of Secondary Heat Sink".

D. Do not stop ECCS pumps. Remain in E-3 and recommence RCS cooldown.			
Proposed Answer:	B		
Explanation (Optional):			
B is correct. Per step 20 of E-3 if RCS pressure is not stable or increasing then transition to ECA-3.1, SGTR With Loss of Reactor Coolant – Subcooled Recovery Desired is directed. A is plausible as it would be the action if SI termination criteria were met. C is plausible since this action is required if heat sink criteria not met for FRPs D is plausible since this action would be performed if PZR level was low.			
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	E-3, Steam Generator Tube Rupture (Rev 42)		
Proposed references to be provided to applicants during examination:			None
Learning Objective:	L1205I09RO		(As available)
Question Source:	Bank #	X	2011 Millstone
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	Last NRC Exam	Millstone 2011 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	(CFR: 43.5)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q79	Tier #		1
	Group #		1
	K/A #	**054** Loss of Main Feedwater (MFW) **2.4.41** Knowledge of the emergency action level thresholds and classifications.	
	Importance Rating		4.6

Proposed Question:

Plant conditions:

- 100% power.
- "B" SEPS engine is out of service for maintenance.
- Loss of offsite power occurs.
- FW-P-37A, TDEFW pump is severely damaged on start.
- FW-P-37B, MDEFW pump is running with no discharge pressure.
- Bus E5 lockout received due to bus fault during transient.
- All SG WR levels are < 65% and lowering.

Which of the following describes the correct response to these conditions? (Reference material provided.)

A. Immediately declare an ALERT due to SA5.

B. Immediately declare a SITE AREA EMERGENCY due to potential loss of two barriers.

C. Attempt to restore FW-P-37B. If after 15 minutes restoration is unsuccessful, declare a SITE AREA EMERGENCY due to potential loss of two barriers.

D. Continue attempts to restore offsite power to Bus E6. If after 15 minutes a second source of power cannot be aligned to Bus E6 then declare an ALERT based on SA5.

Proposed Answer: B

Explanation (Optional):

B is correct. The given plant conditions put Heat Sink CSF in a RED path condition. Electrical status meets EAL requirements for alert (SA5). RED path for Heat Sink meets Potential Loss for both the Fuel Clad and RCS barriers. These two Potential losses meet EAL requirements for FS1. Site Area Emergency should be declared for these conditions as it is the highest EAL exceeded.

Immediate declaration is required as current indications meet EAL, the 15 minute time period is not to be used as a grace period to attempt restoration to prevent declaring and implementing response actions to protect public health and safety.

A is incorrect but plausible. Electrical status meets EAL requirements for Alert (SA5). However conditions are also met for SAE and the higher classification is required.

C is incorrect but plausible. If B EFW pump can be restored and EFW flow established then SAE conditions no longer exist. However, immediate declaration is required as current indications meet EAL and the 15 minute time period is not to be used as a grace period to attempt restoration to prevent declaring and implementing response actions to protect public health and safety.

D is incorrect but plausible. Electrical status meets EAL requirements for Alert (SA5). Attempts to restore power would prevent having to declare the alert. However conditions are also met for SAE and the higher classification is required.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ER-1.1 Classification of Emergencies (Rev 53) F-0.3 Heat Sink Status Tree (Rev 21)

Proposed references to be provided to applicants during examination:				ER-1.1
Learning Objective:	L1509I02SR			(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	(CFR: 43.5)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q80	Tier #		1
	Group #		1
	K/A #	**062** Loss of Nuclear Service Water **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:

- 100% power.
- Service water pump P-41A is tagged out for corrective maintenance.
- Service water pump P-41C trips.
- The Unit Supervisor has entered OS1216.01, "Degraded Ultimate Heat Sink".
- PCCW HX "A" outlet temperature is 90°F and rising slowly.

What action should be taken?

A. Trip the Reactor and enter E-0, continue efforts to restore service water flow.

B. Continue efforts to restore service water flow. If SW cooling cannot be restored before PCCW reaches 100°F, initiate a plant shutdown.

C. Continue efforts to restore service water flow. If SW cooling cannot be restored, initiate a plant shutdown. Shutdown PCCW loads as necessary per Attachment.

D. Continue efforts to restore service water flow. If SW cooling cannot be restored before PCCW reaches 100°F, Trip the Reactor and enter E-0. Shutdown PCCW loads as necessary per Attachment.

Proposed Answer: C

Explanation (Optional):

C is correct. Per OS1216.01 Degraded Ultimate Heat Sink if "A" train PCCW temperature cannot be maintained < 75°F and normal SW flow cannot be established a plant shutdown is initiated and

heat loads on PCCW are secured as necessary.

A is incorrect but plausible. Uncontrolled heat up of the PCCW system could be construed as a loss of PCCW to the RCPs and a reactor trip is required if RCP cooling is lost to two RCPs.

B is incorrect but plausible the temperature limit on PCCW is 100°F. Per OS1012.03 PCCW Loop A Operation precaution 3.13 places a 100°F limit on PCCW temperature to prevent damage to HX

D is incorrect but plausible. Uncontrolled heat up of the PCCW system could be construed as a loss of PCCW to the RCPs and at 100°F a reactor trip is required if RCP cooling is lost to two RCPs.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1216.01, Degraded Ultimate Heat Sink (Rev 22) OS1012.03, Primary Component Cooling Water Loop A Operation (Rev 21)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1193IRO			(As available)
Question Source:	Bank #	X	TEB26694	
	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	(CFR:43.5)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q81	Tier #		1
	Group #		1
	K/A #	**Westinghouse E11** Loss of Emergency Coolant Recirculation **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:

Plant conditions:

- 100% power.
- The plant is 2 days into a 7 day T.S. shutdown action statement due to a breaker problem with RH-P-8A. The breaker and the control switch are the only components on the clearance order.
- A large break LOCA occurs.
- The crew has entered ES-1.3, "Transfer to Cold Leg Recirculation".
- CBS-V-5, "B" train RWST suction isolation fails to close.

Which of the following describes the correct procedure response?

A. Place "B" CBS and "B" RHR pumps in PTL. Continue with ES-1.3.

B. Place both CBS pumps and "B" RHR pump in PTL. Continue with ES-1.3.

C. Place "B" CBS and "B" RHR pumps in PTL. Go to ECA-1.1, "Loss of Emergency Coolant Recirculation".

D. Place both CBS pumps and "B" RHR pump in PTL. Go to ECA-1.1, "Loss of Emergency Coolant Recirculation".

Proposed Answer:	C

Explanation (Optional):

C is correct. Per step 3 of ES-1.3 if CBS-V-5 will not close the 'B' train CBS and RHR pumps are placed in PTL. Per step 4 of ES-1.3 if no RHR pumps are running the direction is given to go to ECA-1.1.

A is incorrect but plausible. Stopping the 'B' RHR and 'B' CBS pumps is correct. Continuation with ES-1.3 would be performed if the 'A' RHR pump were not tagged out. The 'A' CBS pump will remain running and it is plausible that a loss of recirculation will not be recognized or that ES-1.3 will address this by continuing with the procedure.

B is incorrect but it is plausible both CBS pumps will be stopped since a flow path from the RWST will not isolate and there will be no suction supply from RHR pumps. It is plausible that ES-1.3 will address loss of both RHR pump recirculation capability if it is not realized there is an ECA to deal with this condition. It is also plausible the 'A' RHR pump being tagged out will not be recognized and continuation of ES-1.3 is correct.

D is incorrect but it is plausible to stop both CBS pump if not recognized the 'A' CBS pump still has a suction path directly from the containment sump and does not rely on recirculation from the RHR system.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-1.3, Transfer to Cold Leg Recirculation (Rev 28)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1209I01RO			(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	(CFR: 43.5)		
Comments:				

Seabrook Station 2013 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 #	**036** Fuel Handling Incidents **2.2.39** Knowledge of less than or equal to one hour Technical Specification action statements for systems.	
	Importance Rating		4.5

Proposed Question:

Plant conditions:

- "A" Train is protected.
- Core reload is in progress, with 125 assemblies loaded.
- SG secondary manways are removed for sludge lancing
- SG primary side manways are open, with nozzle dams installed, Eddy Current testing is in progress.
- Battery 11D is being replaced.
- Normal battery supply breaker to Bus 11C trips open with battery bus remaining energized from its battery charger.

Which of the following activities is required to be stopped, if any?

A. None.

B. Core reload.

C. SG sludge lancing.

D. Eddy Current testing of the steam generators.

Proposed Answer: B

Explanation (Optional):

B is correct. In MODE 6 T.S. 3.8.3.2 d requires two 125 vdc busses (in the same train) energized from their associated battery banks. Action if not met is to immediately suspend CORE ALTERATIONS.

A is incorrect but plausible. T.S. 3.8.3.2 is not applicable if not in MODE 5 or 6.

C is incorrect but plausible. Sludge lancing could be suspended and manway covers installed to

satisfy containment penetration closure requirements of T.S.3.9.4 D is incorrect but plausible. Eddy current testing could be suspended and containment penetration covers installed to satisfy containment penetration closure requirements of T.S.3.9.4				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S. 3.8.3.2 (Rev 0)			
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			
	55.43	(CFR: 43.2)		
Comments:				

Seabrook Station 2013 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 #	**067**: Plant fire on site **2.4.18** Knowledge of the specific bases for EOPs.	
	Importance Rating		4.0

Proposed Question:

Plant conditions:

- 100% power.
- A fire is reported in "B" Essential Switchgear room by the Fire Brigade Leader.
- The crew enters OS1200.00, "Response to Fire or Fire Alarm Actuation" and transitions to OS1200.01, "Safe Shutdown and Cooldown from the Main Control Room".
- During the transition a fire induced transient generates a Train "B" SI actuation signal.

Which of the following describes the correct course of action?

A. Enter E-0, "Reactor Trip or Safety Injection", Manually actuate SI, when PZR level >95% stop all charging pumps, realign charging to normal, return to OS1200.01.

B. Enter E-0, "Reactor Trip or Safety Injection", Manually actuate SI, transition to ES-1.1, "SI Termination", Return to OS1200.01.

C. Remain in OS1200.01. Implement E-0 actions simultaneously.

D. Remain in OS1200.01. Do not implement E-0.

Proposed Answer: D

Explanation (Optional):

KA match. Knowledge of EOPs bases is required to know when not to implement EOP.

D is correct. Note in OS1200.00 states that E-0 should not be entered for fire induced reactor trip or safety injection should occur. Note in OS1200.01 states that E-0 should not be entered and EOP should only be implemented to mitigate accidents.

A is incorrect. It is plausible to take actions of E-0 to stop high pressure injection of inadvertent SI. However fire conditions warrant remaining in fire abnormal procedures.

B is incorrect. It is plausible to take actions per E-0 and ES-1.1 to address conditions for inadvertent SI. However fire conditions warrant remaining in fire abnormal procedures.

C is incorrect but plausible. Reactor trip and SI require implementing E-0 and immediate actions. These could be taken in parallel with fire abnormal. However fire conditions warrant remaining in fire abnormal procedures.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1200.00 Response to Fire or fire Alarm Actuation (Rev20) OS1200.01 Safe Shutdown and Cooldown From the Main Control Room (Rev19)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8210I09RO			(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	(CFR: 43.5)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q84	Tier #		1
	Group #		2
	K/A #	**Westinghouse E02** SI Termination **EA2.** Ability to determine and interpret the following as they apply to the (SI Termination) **EA2.2** Adherence to appropriate procedures and operation within the limitations in the facility's license and amendments.	
	Importance Rating		4.0

Proposed Question:

Plant conditions:

- SI has occurred due to fault in "D" SG.
- E-2, "Faulted Steam Generator Isolation" has been performed.
- The crew is currently in ES-1.1, "SI Termination".
- Charging has been aligned to the normal charging flow path.
- The crew has stopped both SI pumps.
- The crew has stopped both RHR pumps.

Which condition requires the operator to manually start ECCS pumps?

A. RCS Subcooling decreases to less than 40°F.
B. RCS pressure decreases to less than 1800 psig.
C. "B" SG pressure decreases to less than 585 psig.
D. Containment pressure increases to greater than 4 psig.

Proposed Answer: A

Explanation (Optional):

A is correct. Per ES-1.1 OAS page ECCS Reinitiation Criteria if Subcooling is less than 40°F or PZR level cannot be maintained >7% after completion of step 8 (stopping RHR pumps), then manually align valves and start ECCS pumps as required.

B, C and D are incorrect but plausible as they are the SI manual actuation criteria of E-0.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ES-1.1, SI Termination (Rev 36) E-0, Reactor Trip or Safety Injection (Rev 50)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1226I03RO			(As available)
Question Source:	Bank #			
	Modified Bank#	X	TEB22588	(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			

Content:

	55.43	(CFR: 43.5)

Comments:

TEB22588 Prior to modification:

Plant conditions:

- Inadvertent SI.
- The crew is currently in ES-1.1, "SI Termination".
- At step 7, "Check if SI Pumps Should be Stopped" the crew stopped both SI pumps.
- At step 8, "Check if RHR Pumps Should be Stopped" the crew stopped both RHR pumps.
- Subsequently, one PZR PORV opens and cannot be closed.
- The PORV's block valve cannot be closed.

Which condition requires the operator to manually start ECCS pumps?

A. PZR level decreases to less than 17%.
B. RCS Subcooling decreases to less than 40°F.
C. RCS pressure decreases to less than 1650 psig.
D. Containment pressure increases to greater than 4 psig.

Answer: B

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q85	Tier #		1
	Group #		2
	K/A #	**Westinghouse E08** Pressurized Thermal Shock **2.4.20** Knowledge of the operational implications of EOP warnings, cautions, and notes.	
	Importance Rating		4.3

Proposed Question:

Plant conditions:

- Steam break inside containment occurred at 100% power.
- All systems operated as designed.
- Intact SG NR levels are 15%.
- EFW flow has been throttled to 100 gpm to intact SGs.
- The crew is in FR-P.1, "Response to Imminent Pressurized Thermal Shock Conditions".
- RCS temperature is stable.
- RCS pressure is stable.
- The pressurizer heater control group is energized.
- The crew has just commenced a temperature soak per step 24b.

What action can be performed by the crew within the next hour?

A. Place auxiliary spray in service.
B. Energize additional pressurizer heaters.
C. Increase EFW flow to 300 gpm per SG to raise NR levels to 50%.
D. Place RHR in shutdown cooling mode and commence cooldown to Mode 5.

Proposed Answer:	A

Explanation (Optional):

A is correct. Caution prior to step 17 of FR-P.1 States "An increase in RCS pressure may result in excessive reactor vessel stress. RCS pressure and temperature should be maintained stable while performing subsequent steps in this procedure." Step 24 limits operator actions during the 1 hour soak to things that do not cooldown the RCS or increase RCS pressure. Placing aux spray in service will aid in pressure control and allow operators to lower RCS pressure.

B is incorrect but plausible. Energizing additional pressurizer heaters could be thought of as being performed to generate a pressurizer outsurge to aid in RCS heatup or temperature stabilization.

C is incorrect but plausible. Intact SG levels are usually controlled at 50%. Raising SG to normal control band is plausible.

D is incorrect but plausible. For entry into FR-P.1 RCS temperature has to be < 250°F. This is within the temperature limits of the RHR system operation.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-P.1, Response to Imminent Pressurized Thermal Shock Conditions (Rev 32)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1208I05RO			(As available)
Question Source:	Bank #	X	TEB20663	
	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	(CFR: 43.5)	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q86	Tier #		2
	Group #		1
	K/A #	**005** Residual Heat Removal System (RHRS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.02** Pressure transient protection during cold shutdown	
	Importance Rating		3.7

Proposed Question:

Plant conditions:

- MODE 5, Reduced Inventory, with the RCS intact.
- Both trains of RHR are available, with Train "B" in operation.
- Both PORVs are lined up for LTOP mode of operation.
- Due to indication of RHR leak, the crew enters OS1201.02, "RCS Leak".
- Per the procedure, the crew isolates both trains of RHR by placing both RHR pumps in Pull-to-Lock and closes RHR suction valves RC-V22, V23, V87, and V88.

How does this action affect the Tech. Spec operability of Overpressure Protection Systems?

A. Tech Spec requirements are met. Isolating RHR suction valves does not affect operability of the RHR suction reliefs.

B. Tech Spec requirements are NOT met. At least 1 RHR suction relief must be available for overpressure protection.

C. Tech Spec requirements are met. Both PORVs are available for overpressure control.

D. Tech Spec requirements are NOT met. BOTH PORVs and BOTH RHR suction reliefs are required for overpressure control while in Reduced Inventory.

Proposed Answer:	C	

Explanation (Optional):
C is correct. T.S.3.4.9.3 Mode 5 requirement for overpressure protection is either two PORVs or two RHR suction reliefs or one PORV and one RHR suction relief. With both trains of RHR suction valves isolated neither RHR suction relief is available but both PORVs are available and the LCO is met. A is incorrect but plausible if the RHR suction reliefs are not isolated from the RCS by RC-V22, V23, V87, and V88. B is incorrect but it is plausible that one RHR suction relief is required as at reduced inventory there is no water in the pressurizer. D is incorrect but it is plausible that at reduced inventory both sets of RHR and PORVs relief paths are required, if cooling were lost and RCS pressurized they would be needed to relieve pressure so a low pressure source could inject water.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.4.9.3, Reactor Coolant System, Pressure/Temperature Limits, Overpressure Protection Systems (Amendment No. 115)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8033I14RO			(As available)
Question Source:	Bank #	X	TEB23183	
	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	(CFR: 43.5)	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q87	Tier #		2
	Group #		1
	K/A #	**008** Component Cooling Water System (CCWS) **2.2.40** Ability to apply Technical Specifications for a system.	
	Importance Rating		4.7

Proposed Question:

Plant conditions:

- MODE 3 was entered 24 hours ago following refueling outage.
- RCS pressure 2235 psig.
- RCS temperature 557°F.
- Reactor startup preparations are in progress

After a review of outage work, the PCCW system engineer reports that the head tank level isolation signal to CC-V-426, "A" Train PCCW isolation to WPB will not work. A bistable was incorrectly configured and the retest failed to test CC-V-426 under all required conditions. The T signal input to CC-V-426 was not affected.

Which of the following describes the correct action? (Reference material provided.)

A. PCCW is operable. Startup may continue.

B. Enter T.S. 3.7.3. Repair within 48 hours or be in Cold Shutdown within the following 30 hours. Startup may <u>not</u> continue.

C. Enter T.S 3.7.3. Repair within 72 hours or be in Cold Shutdown within the following 30 hours. Startup may continue.

D. Enter T.S. 3.7.3. Repair within 72 hours or be in Cold Shutdown within the following 30 hours. Startup may <u>not</u> continue.

Proposed Answer:	D	

Explanation (Optional):

D is correct. Basis for T.S.3.7.3 states PCCW system isolations from head tank low levels is required for system operability. With one train of PCCW not operable in mode 3, within 72 hour repair must be made and since already in HOT STANDBY Must be in COLD SHUTDOWN within the following 30 hours.

A is incorrect but plausible. T.S. action applies. CC-V-426 is not discussed in T.S. 3.7.3 surveillance requirements. Surveillance requirements 4.7.3 discusses automatic valves servicing safety related equipment. The WPB has no safety related equipment.

B is incorrect but plausible. T.S. action applies. 48 hour distracter is from the time mode 3 was entered. Time requirement is from point of discovery not from when mode 3 was entered.

C is incorrect but plausible. T.S. action applies but T.S. 3.0.4 will not allow change to mode 2.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S. 3.7.3 (Amendment 32) T.S. 3.7.3 Basis (BC 04-09)

Proposed references to be provided to applicants during examination:				T.S. 3.7.3
Learning Objective:	L8036I15RO, 16RO			(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	(CFR: 43.2)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q88	Tier #		2
	Group #		1
	K/A #	**026** Containment Spray System (CSS) **2.2.37** Ability to determine operability and/or availability of safety related equipment.	
	Importance Rating		4.6

Proposed Question:

Plant conditions:

- MODE 5.
- CBS-P-9A, Containment Building Spray pump impeller has been replaced.

What is required to enter MODE 4? (Reference provided)

A. Perform operational surveillance test prior to MODE 4 entry.

B. Perform a pump head curve verification and operational surveillance test prior to MODE 4.

C. Enter MODE 4, then perform a pump head curve verification and operational surveillance within the next 24 hours.

D. Enter MODE 4 and continue RCS heat up. Test pump using normal quarterly surveillance at the next normal scheduled surveillance interval.

Proposed Answer: B

Explanation (Optional):

B is correct. SSMA 3.5, Post Maintenance Testing fig 5.4, 7. Pumps, requires pump curve verification and an operational test for OPERABILITY. T.S. 3.6.2.1 requires 2 CBS systems OPERABLE in Mode 4. The retests are required prior to Mode 4 entry.

A is incorrect but plausible as an operability test is required prior to Mode 4 entry.

C is incorrect but plausible. Some Tech Specs such as 3.7.1.1 allow mode entry with inoperable equipment for up to 24 hours to allow post-maintenance testing.

D is incorrect but plausible. If it is discovered a surveillance has been missed, T.S. 4.0.3 allows delaying the surveillance up to 24 hours or up to the limit of the specified surveillance interval which ever is greater.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S. 3.6.2.1, Containment Systems, Depressuization and Cooling Systems, Containment Spray System (Amendment No. 128) T.S.4.0.3, Limiting Condition for Operation (Amendment No. 114) T.S.3.7.1.1, Plant Systems, Turbine Cycle, Safety Valves (Rev 0) SSMA 3.5, Post Maintenance Testing (Rev 13)			
Proposed references to be provided to applicants during examination:				SSMA 3.5 Pg 1-13. Fig 5.2, Fig 5.3, Fig 5.4 sh 1-14
Learning Objective:	L1514I19SR			(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	(CFR: 43.5)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q89	Tier #		2
	Group #		1
	K/A #	**076** Service Water System (SWS) **2.4.30** Knowledge of events related to system operation/status that must be reported to internal organizations or external agencies, such as the State, the NRC, or the transmission system operator.	
	Importance Rating		4.1

Proposed Question:

Plant Conditions:

- 100% power
- The following maintenance is in progress on "A" train Service Water.
  - SW-P-41A motor replacement.
  - SW-V-2, "SW-P-41-A discharge valve" MOV diagnostics.

Which of the following situations requires immediate state and/or federal notification?

A. 10 gallon oil spill to the SW fore-bay.

B. 10 gallon oil spill to asphalt outside the SWPH door.

C. Maintenance tech injury requiring immediate transport and hospitalization.

D. Operating permit holder connects to and closes SW-V-22, "SW-P-41-C discharge valve".

Proposed Answer: A

Explanation (Optional):

A is correct. ON1244.01, Spill Response Attachment 'B' Requires NHDES, US Coast Guard and National Response Center notification of <u>any</u> petroleum spill with a potential to discharge to the ocean.

B is incorrect but plausible. ON1244.01, Spill Response Attachment 'B' a spill of < 25 gallons of oil to land where it will ultimately seep into ground water is reportable within 1 hour

C is incorrect but plausible. An injury requiring transport to the hospital is an OSHA reportable event. An incident involving hospitalization of 3 or more employees is required to be reported to OSHA within 8 hours.

D is incorrect but plausible. Inadvertent closure of SW-V-22 on the running SW pump will case a Tower Actuation signal. This would require a 4 hour report t the NRC as a System Actuation.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	NARC 3.3.1 Synopsis of Immediate Reporting Requirements (Rev 146) ON1244.01, Spill Response (Rev 24)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1522I01RO			(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	(CFR: 43.5)	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q90	Tier #		2
	Group #		1
	K/A #	**078** Instrument Air System (IAS) **A2** Ability to (a) predict the impacts of the following malfunctions or operations on the IAS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: **A2.01** Air dryer and filter malfunctions	
	Importance Rating		2.9

Proposed Question:

Plant conditions:

- 100% power.
- Containment IA dryer IA-D-2A develops an internal air leak.
- D4985, CONTM INSTRUMENT AIR HDR A PRESS LOW goes into alarm.
- IA-PI-8024, "A" Containment IA header Pressure indicates 95 psig and decreasing.
- ON1242.02, "Loss of Containment Instrument Air" is entered.
- IA-V-530, IA to Containment Isolation valve has been opened.
- "A" Containment IA header pressure continues to decrease.

Which of the following describes the correct actions to be taken?

A. Monitor Containment PCCW isolation valve positions.

B. Trip the reactor, Go to E-0, "Reactor Trip or Safety Injection". Stop all RCPs.

C. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". Stop "A" and "D" RCPs when immediate actions are complete.

D. Trip the reactor. Go to E-0, "Reactor Trip or Safety Injection". Stop "B" and "C" RCPs when immediate actions are complete.

Proposed Answer: A

Explanation (Optional):

A is correct. The design of containment instrument air system is such that when cross connected with the instrument air system by opening IA-V-530 the pressure instrument for the failed loop is

isolated by a check valve. There is a note prior to step 1 of ON1242.02 that reminds the operator of this when the procedure is entered. Also containment isolation AOVs inside containment are supplied from both loops of containment instrument air. Even if the 'A' loop depressurized completely valves should remain open. PCCW isolation valves are monitored in the procedure to ensure this occurs and if not a reactor trip is required.

B is incorrect but plausible. An "A" train Phase B isolation signal closes valves in both loops of PCCW supply to containment and would require stopping all RCPs. Loss of instrument air loop 'A' inside containment could be the only supply to 1 valve in each PCCW loop causing loss of cooling flow to all RCPS.

C is incorrect but plausible. Instrument air loop 'A' inside containment could be the only supply to the "A" loop PCCW containment isolation valves IRC. Loss of this supply could result in the loss of cooling to RCPs A and D. There is a common misconception as to which RCPs are cooled by which loop of PCCW.

D is incorrect but plausible. Instrument air loop 'A' inside containment could be the only supply to the "A" loop PCCW containment isolation valves IRC. Loss of this supply could result in the loss of cooling to RCPs B and C. There is a common misconception as to which RCPs are cooled by which loop of PCCW.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1242.02, Loss of Containment Instrument Air (Rev 12) PID-1-IA-B20643, Instrument Air Containment Building Detail (Rev 13)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1194I06RO			(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	(CFR: 43.5)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q91	Tier #		2
	Group #		2
	K/A #	**028** Hydrogen Recombiner and Purge Control System (HRPS) **2.1.20** Ability to interpret and execute procedure steps.	
	Importance Rating		4.6

Proposed Question:

Plant conditions:

- 100% power.
- Large break LOCA occurred causing SI and entry into E-0, "Reactor Trip or Safety Injection".
- After exiting E-0 the crew enters FR-Z.1, "Response to Containment High Pressure" based on high containment pressure.
- Step 6 "Check Hydrogen Concentration" is being performed.

When would the Hydrogen Recombiner be required to be placed in service?

A. Approximately 9 hours following a large break LOCA.

B. Following any reactor vessel head venting evolutions.

C. When Containment hydrogen concentration reaches 6%.

D. When Containment hydrogen concentration reaches 0.5%.

Proposed Answer: D

Explanation (Optional):

D is correct. FR-Z.1, Response to Containment High Pressure places the hydrogen monitor in service, if hydrogen concentration is < 4.0% and >0.5% the hydrogen recombiners are placed in service.

A is incorrect but plausible. UFSAR figure 6.2-96 graph of Containment Hydrogen Concentration with and without Recombiner shows that containment hydrogen concentration reaches ~0.5%at

approximately 9 hours. Refer to to UFSAR 6.2.5 Combustible Gas Control.

B is incorrect but plausible. FR-I.3 places containment hydrogen analyzers inservice and checks containment hydrogen prior to and during reactor head venting. The recombiner is not placed in service unless concentration reaches 3%. Calculations are peformed prior to venting to lilmit containment hydrogen increase to < 3%.

C is incorrect but plausible as this is the concentration that hydrogen becomes explosive. Procedure does not allow recombiners to be operated >4%. It could be misconstrued that starting recombiner is required at 6%.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	FR-Z.1, Response to Containment High Pressure (Rev 23) UFSAR section 6.2.5 Combustible Gas Control in Containment (Rev 15) UFSAR Figure 6.2-96, Containment Hydrogen Concentration with and without Recombiner (Rev 0) FR-I.3, Response to Voids in Reactor Vessel (Rev 26)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1212I07RO			(As available)
Question Source:	Bank #	X	TEB25488	
	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		
	55.43	(CFR: 43.5)	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q92	Tier #		2
	Group #		2
	K/A #	**034** Fuel Handling Equipment System (FHES) **K4** Knowledge of design feature(s) and/or interlock(s) which provide for the following: **K4.03** Overload protection	
	Importance Rating		3.3

Proposed Question:

Plant conditions:

- Defueled.
- A Master Lee Debris Canister has been moved from containment to the SFP using the Fuel Transfer System.
- The container has been loaded to 2500 lbs.

What condition will allow the Spent Fuel Crane to move this basket, if at all?

A. This Load cannot be moved with the Spent Fuel Crane.

B. Move Spent Fuel Crane over upender. Latch onto basket with Spent Fuel handling tool. Go to Raise on the hoist joy stick.

C. Move Spent Fuel Crane over upender. Latch onto basket with Spent Fuel handling tool. Press and hold the Hoist Load Bypass pushbutton. Go to Raise on the hoist joy stick.

D. Depress and hold Travel Override until it latches in. Move Spent Fuel Crane over upender. Latch onto canister with Spent Fuel handling tool. Press and hold Hoist Load Bypass pushbutton. Go to Raise on the hoist joy stick.

Proposed Answer: A

Explanation (Optional):

A is correct. OX1415.04 describe the surveillance performed on the spent fuel bridge to verify crane overload setpoints to meet requirements of TR 27-3.9.7. The hoist overload cutout is verified to prevent lifting loads greater than 2500 pounds and is actual verified that it cannot lift the 2474 lb test weight in the surveillance.

B is incorrect but it is plausible as the directions do get the spent fuel crane in position to lift the load as in the normal operating procedure for crane operation. If the hoist overload setpoints are not known the crane could lift this weight. C is incorrect but it is plausible that using Hoist Load Bypass the crane could lift this load. D is incorrect but it is plausible that using Travel override and Hoist load bypass this load could be lifted.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OS1015.07, Spent Fuel Bridge Assembly Operation (Rev 21) OX1415.04, Spent Fuel Bridge Assembly Weekly Operational Test, (Rev 13) T.R.27-3.9.7 Crane Travel- Spent Fuel Storage Areas (Rev 112)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8060I05RO			(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	(CFR: 41.7)	
	55.43	(CFR:43.7)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q93	Tier #		2
	Group #		2
	K/A #	**072** Area Radiation Monitoring (ARM) System **2.2.42** Ability to recognize system parameters that are entry-level conditions for Technical Specifications.	
	Importance Rating		4.6

Proposed Question:

Plant condition:

- 100% power.
- A part 21 issue has been identified by engineering for the Post LOCA radiation monitors.
- The SM has determined that both Post LOCA radiation monitors are inoperable.
- Replacement parts have a 3 week lead time.

Which of the following is the correct action? (Reference provided)

A. Enter T.S. 3.0.3 due to both channels being inoperable. Go to MODE 4 per T.S. 3.0.3.

B. Enter T.S. 3.0.3 due to both channels being inoperable. Go to MODE 5 per T.S. 3.0.3.

C. Enter T.S. 3.3.3.6. Submit special report to commission within 14 days. Initiate alternate method for monitoring.

D. Enter T.S. 3.3.3.6. Submit special report to commission within 44 days. Initiate alternate method for monitoring.

Proposed Answer: C

Explanation (Optional):

C is correct. With Less than the minimum channels operable action c applies. An alternate method of monitoring is required to be initiated and if not repaired in 7 days a report to the NRC is required within 14 days.

A is incorrect but it is plausible that if no POST LOCA monitors are available then T.S. 3.0.3 would be entered reduce mode to mode 4 where the T.S. 3.3.3.6 is not applicable.

B is incorrect but it is plausible that if no POST LOCA monitors are available then T.S. 3.0.3 would be entered reduce mode to mode 5 where the T.S. 3.0.3 is not applicable.

D is incorrect but it is plausible that action a could apply and if not repaired in 30 days then a report is due to the NRC within the next 14 days.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.3.3.6, Instrumentation, Monitoring Instrumentation, Accident Monitoring Instrumentation (Amendment No. 114) T.S.3.0.3, Limiting Conditions for Operation (Amendment No.114)

Proposed references to be provided to applicants during examination:				T.S3.3.3.6
Learning Objective:	L8059I14RO			(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	(CFR: 43.2)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q94	Tier #		3
	Group #		
	K/A #	**2.1.4** Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.	
	Importance Rating		3.8

Proposed Question:

Plant conditions:

- 100% power.
- An on-shift Fire Brigade member becomes ill and must be taken to the hospital.
- There are 5 hours left until shift change.

What action is required?

A. Action must be taken to obtain a replacement fire brigade member within two hours.
B. Action must be taken to obtain a replacement fire brigade member within four hours.
C. None. The vacant fire brigade position can remain unmanned until shift turnover.
D. A replacement fire brigade member must be on-shift within two hours AFTER the scheduled shift turnover.

Proposed Answer: A

Explanation (Optional):

A is correct. Per OPMM, Chapter 2, Shift Composition, Section 1.2, Shift Compliment contains the following NOTE:
The shift crew composition, including a Health Physics Technician and the Fire Brigade may be one less than the minimum requirements of Table 6.2-1 of Technical Specifications for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to fill the required position. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crew member being late or absent.

D is incorrect but plausible. There is a time limit for replacement of the brigade member, however it is from the time of incident vice time from turnover.

C is incorrect but plausible. There is a time limit for replacement of the brigade member,

however it is 2 hours vice 4 hours.

B is incorrect but plausible. There is a time allowance for replacement, however it is 2 hours vice "until turnover" which is 5 hours away.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.6.2.2 Station Staff (Amendment 124,104) OPMM Ch 2.1 Shift composition (Rev 83)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1505I02RO			(As available)
Question Source:	Bank #	X	TEB30036	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2007 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		
	55.43	(CFR: 43.2)	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
Q95	Tier #		3
	Group #		
	K/A #	**2.1.41** Knowledge of the refueling process.	
	Importance Rating		3.7

Proposed Question:

Given the following:

Date	Time	Activity
12/30/2011	2200	Enters MODE 2.
12/31/2011	0000	Enters MODE 3.
12/31/2011	1320	Enters MODE 4.
12/31/2011	2210	Enters MODE 5.
01/01/2012	2200	The first Reactor Vessel Head Stud is detensioned.
01/03/2012	0100	The Reactor Vessel Head is removed.

Which ONE of the following is (1) the EARLIEST time to commence fuel movement in accordance with Technical Specifications, and (2) the basis for the time requirement?

A. (1) 01/03/12 at 0600

(2) Ensures the heat load assumptions specified in the safety analysis are met to prevent boiling in the Spent Fuel Pool.

B. (1) 01/03/12 at 0600

(2) Ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis.

C. (1) 01/03/12 at 0800

(2) Ensures the heat load assumptions specified in the safety analysis are met to prevent boiling in the Spent Fuel Pool.

D. (1) 01/03/12 at 0800

(2) Ensures that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the values specified in the safety analysis.

Proposed Answer:	D	

Explanation (Optional):

D is correct. Per T.S.3.9.3 The reactor must be subcritical for at least 80 hours to move fuel. From the conditions given in the stem of the question this is the time mode 3 was entered. This is also captured in OS1000.09, step 4.2.4. The basis for 80 hour delay time is to ensure sufficient time has elapsed to allow the radioactive decay of short lived fission products.

A is incorrect but is plausible the 80 hour limit starts from entry into Mode 2 and the basis for the delay could be to allow reduction of heat generation and the heat load on the spent fuel pool. OS1000.09 Precaution 3.20 discusses spent fuel pool temperatures with various combinations of SW.

B is incorrect but is plausible the 80 hour limit starts from entry into Mode 2 and the basis for the delay is to ensure sufficient time has elapsed to allow the radioactive decay of short lived fission products.

C is incorrect but it is plausible as the time is based correctly on entry into mode 3 and the basis for the delay could be to allow reduction of heat generation and the heat load on the spent fuel pool. OS1000.09 Precaution 3.20 discusses spent fuel pool temperatures with various combinations of SW.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.9.3, Refueling Operations, Decay Time (Amendment No. 95) T.S.3.9.3 Basis Decay Time (Amendment No. 93) OS1000.09 Refueling Operation Step 4.2.4 (Rev 25)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1179I03RO			(As available)
Question Source:	Bank #	X	2011 Turkey Point	
	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	(CFR: 43.6)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q96	Tier #		3
	Group #		
	K/A #	**2.2.36** Ability to analyze the effect of maintenance activities, such as degraded power sources, on the status of limiting conditions for operations.	
	Importance Rating		4.2

Proposed Question:

With the plant at 100% power Operations removes the "A" EDG from service for a scheduled maintenance outage.

What action(s) must be taken as a result?

A. Demonstrate the operability of the "B EDG within 8 hours by starting and verifying that the engine reaches rated speed and voltage in 10 seconds. Verify two offsite sources available within one hour.

B. Demonstrate the operability of the "B" EDG within 8 hours. A modified start involving idling and graded acceleration may be used. Verify two offsite sources available within one hour.

C. Verify two offsite sources operable within one hour and every 7 days thereafter.

D. Verify two offsite sources operable within one hour and at least once every 8 hours thereafter.

Proposed Answer: D

Explanation (Optional):

D is correct. Removing the "A" EDG from service requires entry into 3.8.1.1 action b. Action b. 1) requires verifying remaining sources within 1 hour and each 8 hours after. The OPERABILITY of the remaining Diesel is not required if the inoperable diesel was declared inoperable due to preplanned maintenance.

A is incorrect but plausible as the 8 hour requirement is a requirement of T.S.3.8.1.1 action c. 1) and the speed and voltage in 10 seconds is part of the 184 day surveillance 4.8.1.1.2 e.

B is incorrect but it is plausible as this is a requirement of T.S.3.8.1.1 action c. 1) if one offsite

source and one diesel generator are inoperable. C is incorrect but it is plausible as the normal frequency for the offsite verification is 7 days.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	T.S.3.8.1.1, Electrical Power Systems, A.C. Sources, Operating (Amendments 114, 98, 97, 80)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L8020I27RO			(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	(CFR: 43.2)		
Comments:				

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q97	Tier #		3
	Group #		
	K/A #	**2.3.5** Ability to use 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 conditions:

- The plant is at 100% power.
- RM-6504, WG Compressor Discharge Activity radiation monitor becomes inoperable.

What is the required action? (Reference material provided.)

A. Release path via this pathway may not continue until RM-6504 is operable.

B. Effluent releases via this pathway may continue. RM-6503, Waste Gas Compressor Inlet Radiation Monitor may be used instead of collecting grab samples.

C. Effluent releases via this pathway may continue. RM-6503, Waste Gas Compressor Inlet Radiation Monitor is a redundant backup to RM-6504. The associated action statement need not be entered.

D. Effluent releases via this pathway may continue. RM-6503, Waste Gas Compressor Inlet Radiation Monitor may be used as an alternate to RM-6504 and a grab sample must be collected and analyzed within 24 hours.

Proposed Answer: B

Explanation (Optional):

B is correct. With RM-6504 inoperable ODCM C.5.2 action 33 allows release to continue provided grab samples are taken once per 12 hours. Action 33 also allows RM-6503 to be used instead of taking grab samples.

A is incorrect but it is plausible that when the monitor that isolates the release path is inoperable, release from that path may not continue.

C is incorrect but it is plausible the monitor in series with RM-6504 could be a redundant backup and no actions are required.

D is incorrect but it is plausible that with RM-6503 in service the grab sample requirements could be extended to 24 hours.				
Technical Reference(s): (Attach if not previously provided) (including version/revision number)	Offsite Dose Calculation Manual, Part A, Section 5.2, Radioactive Gaseous Effluent Monitoring Instrumentation (Rev 34)			
Proposed references to be provided to applicants during examination:				ODCM Part A Section 5.2
Learning Objective:	L8064I05RO			(As available)
Question Source:	Bank #	X	TEB19978	
	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	(CFR: 43.4)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q98	Tier #		3
	Group #		
	K/A #	**2.3.11** Ability to control radiation releases.	
	Importance Rating		4.3

Proposed Question:

Plant conditions:

- 100% power.
- Waste Test Tank (WTT) "A" liquid effluent waste (LEW) permit sample taken at 0200 3/1/13.
- WTT "A" permit issued at 0400 3/1/13.
- WTT "A" discharge commenced at 0600 3/1/13.

Subsequently at 2330 on 3/1/13:

- CW-P-39A tripped.
- Crew entered ON1238.01, Circulating Water System Malfunction and stabilized the plant.

Currently it is 0300 3/2/13

Which of the following describes the correct action to take regarding the WTT discharge?

A. Reduce WTT discharge flow to 2/3 of the original permit value.

B. Stop the release. The WTT must be placed on recirc and a new LEW permit requested.

C. Stop the release. Discharge may be resumed if CW-P-39A is returned to service by 0600, 3/2/13.

D. Maintain current WTT discharge flow. LEW permits are always based on only one CW pump in operation.

Proposed Answer:	B

Explanation (Optional):

B is correct. Precaution 3.6 in ON1018.08 requires stopping release if tunnel flow is reduced. The LEW permit is limited to 24 hours from sample collection if the release is stopped. Current time is beyond 24 hours. An additional sample must be obtained to restart the release.

A is incorrect but plausible. Loss of 1 CW pump has reduced tunnel flow to 2/3 of what the LEW permit is based on. It is plausible that reducing release flow could be allowed. However, precaution 3.6 in ON1018.08 requires stopping release if tunnel flow is reduced.

C is incorrect but plausible. Current time is <24 hours from release start. It is plausible the time limit for the release is from release start. Time limit is from sample collection not release start. 24 hours from sample has expired.

D is incorrect but plausible. LEW release permits at 100% power are based on 3 CW/1SW pump operation not 1CW/1SW pump operation. The release must be stopped.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ON1018.08 WTT A Discharge to Transition Structure (Rev10) CX0917.01 Liquid Effluent Releases (Rev20)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1517I06SR			(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	(CFR: 43.4)	

Comments:

Seabrook Station 2013 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 #	**2.4.5** Knowledge of the organization of the operating procedures network for normal, abnormal, and emergency evolutions.	
	Importance Rating		4.3

Proposed Question:

Plant conditions:

- Reactor Trip with Safety Injection.
- Crew has transitioned to E-1, "Loss Of Reactor or Secondary Coolant"
- Critical Safety Function (CSF) status as follows:
  - Valid Core Cooling ORANGE Path (C.2)
- The crew is performing FR-C.2
- Critical Safety Function (CSF) status update as follows:
  - Valid Heat Sink RED Path (H.1)
  - Valid Containment RED Path (Z.1)

What action should be taken?

A. Immediately transition to FR-H.1.
B. Immediately transition to FR-Z.1.
C. Complete the actions of FR-C.2 and immediately transition to FR-H.1.
D. Complete the actions of FR-C.2 and immediately transition to FR-Z.1

Proposed Answer: A

Explanation (Optional):

A is correct. Per OP 9.2, Emergency Operators Users Guide, Section 4.3, Control Room Usage of Status Trees, the order of priority for critical safety functions is Subcriticality, Core Cooling, Heat Sink, Integrity, Containment, Inventory, Emergency Recirculation, and RDMS. The order of severity priority is Red, Orange, Yellow, and Green. If any Orange terminus is encountered, the operator is expected to monitor all of the remaining trees, if no Red terminus is present, then

suspend any ERP or ECA and address the Orange condition. If during the performance of an Orange condition FRP, any Red condition or higher priority Orange arises, then the higher priority condition should be addressed and the original Orange FRP is suspended.

B is incorrect but it is plausible that Containment is a higher priority than Heat Sink.

C is incorrect but it is plausible to continue with Core Cooling as it is a higher priority function that either Heat Sink or Containment.

D is incorrect but it is plausible to continue with Core Cooling as it is a higher priority function that either Heat Sink or Containment.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	OP 9.2, Transient Response Procedure User's Guide (Rev 16)

Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1196I06SR			(As available)
Question Source:	Bank #	X	TEB26546	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2007 Seabrook NRC		

(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	(CFR: 43.5)	

Comments:

Seabrook Station 2013 Licensed Operator NRC Written Exam
ES-401-5 Written Examination Question Worksheet

Examination Outline Cross-reference:	Level	RO	SRO
Q100	Tier #		3
	Group #		
	K/A #	**2.4.40** Knowledge of SRO responsibilities in emergency plan implementation.	
	Importance Rating		4.5

Proposed Question:

Which Short Term Emergency Director (STED) responsibility can be delegated?

A. Approval of Protective Action Recommendations.

B. Authorization of emergency radiation exposures.

C. Decision and direction to notify offsite authorities.

D. Notifying plant personnel of E-PLAN implementation.

Proposed Answer: D

Explanation (Optional):

D is correct. Notifying plant personnel is a responsibility that can be delegated. This action is typically performed by the Work Control Supervisor in the process of completing the associated ER1.2 Event Checklists for the applicable emergency plan classification level.

A is incorrect but plausible. Determination of PARs for SAE and GE is typically performed by the Work Control Supervisor while performing ER-1.2 checklists. Once the WCS has the PARs actions completed on the checklist the STED must then approve them. ER1.2, Section 2.0, Responsibilities specifically states that this responsibility cannot be delegated by the STED

B is incorrect but plausible. Coordination of actions in the field is normally performed from the OSC and involves Health Physics support. It is plausible that Health Physics would have the authority to authorize emergency radiation exposures, however ER1.2, Section 2.0, Responsibilities specifically states that this responsibility cannot be delegated.

C is incorrect but plausible. The Work Control Supervisor makes the initial notifications to offsite authorities, however this process is directed to be performed by the STED.

Technical Reference(s): (Attach if not previously provided) (including version/revision number)	ER-1.2, Emergency Plan Activation (Rev 61)			
Proposed references to be provided to applicants during examination:				None
Learning Objective:	L1509ISR			(As available)
Question Source:	Bank #	X	TEB20482	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	Last NRC Exam	2009 NRC Remediation 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	(CFR: 43.5)		
Comments:				