

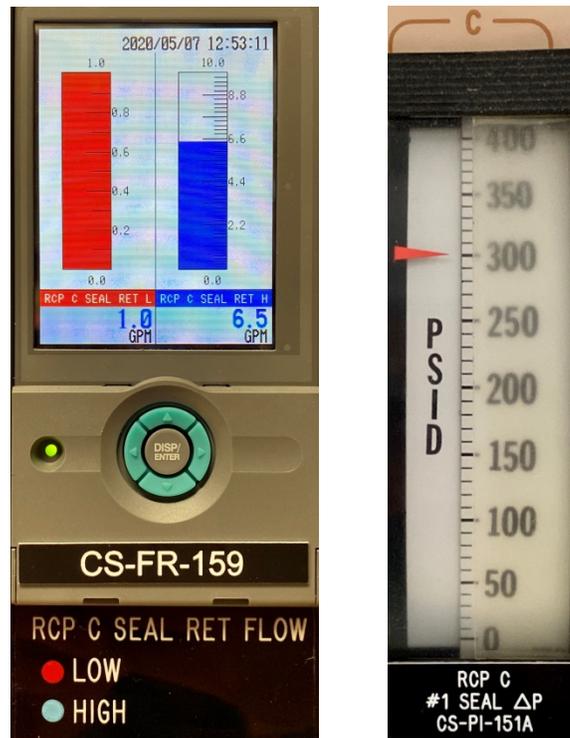
Examination Outline Cross-reference:	Level	RO	SRO
Q1	Tier #	1	
	Group #	1	
	K/A #	000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1 EK1.05 Knowledge of the operational implications and/or cause and effect relationships of the following as they apply to a Reactor Trip: Decay power as a function of time.	
	Importance Rating	3.3	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The crew is performing the actions of FR-S.1, “Response to Nuclear Power Generation/ATWS” initiated by a Loss of All Feedwater event. • Power Range NI channels are 6%. • The crew is checking SG levels. • All SG NR levels are off-scale low. • All SG WR levels are approximately 62%. • C0722 – “Total EFW Flow” is 750 gpm. <p>Based on these conditions, what action will the crew take to control EFW flow?</p> <p>A. Maintain current EFW flow until SG level is greater than 65% WR in at least two steam generators.</p> <p>B. Maintain current EFW flow until SG level is greater than 15% NR in at least one steam generator.</p> <p>C. Increase total EFW flow to greater than 880 gpm until SG level is greater than 65% WR in all steam generators.</p> <p>D. Increase total EFW flow to greater than 880 gpm until SG level is greater than 6% NR in at least one steam generator.</p>			

Proposed Answer:	D.		
Explanation (Optional):			
<p>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 and remove initial decay heat load.</p> <p>C is incorrect but plausible. Flow requirement is correct however, only one SG need be recovered, not all.</p> <p>A is incorrect but plausible. This is criteria to maintain heat sink in E-1 not FR-S.1.</p> <p>B is incorrect but plausible. This is criteria to maintain heat sink in E-1 with containment adverse.</p>			
Technical Reference(s):	FR-S.1, "Response to Nuclear Power Generation/ATWS" Rev 30.		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	L1200I13RO		
Question Source:	Bank #	X	TEB 34973
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2013 Seabrook NRC Exam 2009 Comanche Peak NRC Exam (same K/A)		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(8), (10)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q2		Tier #	1	
		Group #	1	
		K/A #	000009 (EPE 9) Small Break LOCA / 3 EK2.03 Knowledge of the interrelations between the small break LOCA and the following: S/Gs	
		Importance Rating	3.0	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A LOCA has occurred. • Containment pressure is 10 psig and increasing. • RCS pressure is 1400 psig and stable. • The crew is implementing ES-1.2, "Post LOCA Cooldown and Depressurization". <p>Which of the following identifies the method that the crew will use to cooldown the RCS?</p> <p>A. Steam Dumps at the maximum rate. B. Steam Dumps at less than 100 °F/hr. C. ASDVs at the maximum rate. D. ASDVs at less than 100 °F/hr.</p>				
Proposed Answer:		D.		
Explanation (Optional):				
D is correct. Step 8 of ES-1.2 will direct the crew to initiate cooldown to cold shutdown at a rate of <100 °F/hr. The condenser steam dumps are unavailable because containment pressure exceeded 4 psig, causing a main steam line isolation. The crew must use the ASDVs to perform				

the cooldown.			
A is incorrect but plausible. The student would choose this answer if they failed to identify that the MISVs would be shut, precluding use of the condenser steam dump valves. Other procedures such as E-3 utilize a maximum cooldown rate.			
B is incorrect but plausible. The student would choose this answer if they failed to identify that the MISVs would be shut, precluding use of the condenser steam dump valves.			
C is incorrect but plausible. The student would choose this answer if they failed to understand the limitations of the cooldown rate in ES-1.2 vs other procedures such as E-3.			
Technical Reference(s):		ES-1.2, "Post LOCA Cooldown and Depressurization" Rev 40.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1204I03		
Question Source:	Bank #	X	14253
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:		Robinson 2011 NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q3		Tier #	1	
		Group #	1	
		K/A #	000015 (APE 15) Reactor Coolant Pump Malfunctions / 4 2.1.25 Ability to interpret reference materials such as graphs, monographs and tables which contain performance data.	
		Importance Rating	3.9	
Proposed Question:				
Plant conditions: <ul style="list-style-type: none"> • The 'C' RCP has been started in support of plant start up. • D4604, "RCP C No. 1 Seal Leak Off Flow High" alarms. • 'C' RCP seal dp and #1 seal leak rate are as shown below: 				



What is the status of seal leakoff flow and what action is required in accordance with OS1201.01, "RCP Malfunction"?

(reference provided)

- A. Seal leak off flow is in the normal operating range. Check No. 2 seal leak off flow.
- B. Seal leak off flow is in the normal operating range. Check seal water inlet temperature is less than 230 °F.
- C. Seal leak off flow is in the prohibited operating range. Verify reactor trip breakers are open and stop the affected RCP.
- D. Seal leak off flow is in the prohibited operating range. Increase RCS pressure to support continued operation of the 'C' RCP.

Proposed Answer:	C.	
Explanation (Optional):		
C is correct. For the given plant conditions, seal dp and #1 seal leak off flow is in the prohibited		

<p>operating range. Step 4 RNO of OS1201.01 requires the RCP to be shut down after the trip breakers are verified open.</p> <p>A is incorrect but plausible. If the seal dp and #1 seal leak off flow were in the normal operating range, step 4 RNO of OS1201.01 would then check #2 seal leak off flow via step 9. If the student is unable to correctly interpret the graph this would be a possible answer.</p> <p>B is incorrect but plausible. If the student is unable to correctly interpret the graph this would be a possible answer. If RCP seal are in the normal operating band, this is the next step in the procedure, step 5.</p> <p>D is incorrect but plausible. The plant conditions as given are during a plant startup. It is a common misconception that performance issues associated with seal dp and seal leak off flow could be mitigated by adjusting RCS pressure which is under the control of the operator at this time. This would be incorrect as OS1201.01 requires that the RCP be shut down.</p>			
<p>Technical Reference(s):</p>		<p>OS1201.01, "RCP Malfunction" Rev 19.</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>OS1201.01 Attachment D</p>
<p>Learning Objective:</p>	<p>SBK LOP L1181I 03</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		
	<p>Comprehension or Analysis</p>		<p>x</p>
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:		Level	RO	SRO
Q4		Tier #	1	
		Group #	1	
		K/A #	000025 (APE 25) Loss of Residual Heat Removal System / 4 AA2.01 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Proper amperage of running LPI/decay heat removal/RHR pump(s)	
		Importance Rating	2.7	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Mode 5. • Reactor vessel level is minus 18 inches. • The 'A' RHR pump is in standby. • The 'B' RHR pump motor current, discharge pressure and flow are fluctuating. • The crew is implementing OS1213.01, "Loss of RHR During Shutdown Cooling". <p>What is the first action the crew will take in OS1213.01?</p> <p>A. Transition to OS1213.02, "Loss of RHR While at Reduced Inventory or Midloop Conditions".</p> <p>B. Place the control switch for <u>ONLY</u> the 'B' RHR pump in Pull to lock.</p> <p>C. Place the control switches for <u>BOTH</u> RHR pumps in Pull to Lock.</p> <p>D. Start the 'A' RHR pump.</p>				
Proposed Answer:		C.		

Explanation (Optional):			
<p>C is correct. OS1213.01, "Loss of RHR During Shutdown Cooling", requires that both RHR pumps be placed in PTL when one show signs of cavitation. The standby pump will be started later in the procedure.</p> <p>A is incorrect but plausible. OS1213.01, "Loss of RHR During Shutdown Cooling" will direct transition to OS1213.02, "Loss of RHR While at Reduced Inventory or Midloop Conditions" only if reactor vessel level is not above -36 inches.</p> <p>B is incorrect but plausible. Initially both RHR pumps are placed in PTL. Later in the procedure, the standby will be started. It is a common misconception that the standby pump will not be placed in PTL. This is required for equipment protection.</p> <p>D is incorrect but plausible. The standby RHR pump will be started in this procedure, but not before its control switch is placed in PTL and conditions are established to start it.</p>			
Technical Reference(s):		OS1213.01, "Loss of RHR During Shutdown Cooling" Rev 20.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1705I 06		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	(10)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q5		Tier #	1	
		Group #	1	
		K/A #	000026 (APE 26) Loss of Component Cooling Water / 8 AA1.01 Ability to operate and / or monitor the following as they apply to the Loss of Component Cooling Water: CCW temperature indications	
		Importance Rating	3.1	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% Power. • 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. • The crew has entered OS1212.01, "PCCW System Malfunction". <p>Which of the following is the cause of these indications?</p> <p>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.</p>				
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 HX 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 HX 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):	OS1212.01, "PCCW System Malfunction" Rev 14		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8036I 04		
Question Source:	Bank #	x	TEB 34960
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2013 Seabrook NRC Exam		

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

Examination Outline Cross-reference:	Level	RO	SRO
Q6	Tier #	1	
	Group #	1	
	K/A #	000027 (APE 27) Pressurizer Pressure Control System Malfunction / 3 2.4.50 Ability to verify system alarm setpoints and operate controls identified in the alarm response manual.	
	Importance Rating	4.2	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 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 <u>setpoint</u> drifts from 2235 psig to 2160 psig, and components reposition. <p>After placing the Pressurizer Master Pressure Controller to MANUAL, what action will the Reactor Operator take with the Master Pressurizer Pressure Controller in accordance with the alarm response procedure?</p> <p>A. Operate the INCREASE pushbutton, which will energize the control group and available backup heaters and close both spray valves ONLY.</p> <p>B. Operate the DECREASE pushbutton, which will energize the control group and available backup heaters and close both spray valves ONLY.</p> <p>C. Operate the INCREASE pushbutton, which will energize the control group and available backup heaters and close both spray valves and the 'A' PORV.</p> <p>D. Operate the DECREASE pushbutton, which will energize the control group and available backup heaters and close both spray valves and the 'A' PORV.</p>			

E.		
Proposed Answer:	B.	
Explanation (Optional):		
<p>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.</p> <p>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.</p> <p>C is incorrect but plausible. Operation of the INCREASE push button will raise controller output and cause the spray valves to open. This is the opposite of what should be done for the above description of events in the stem of the question. Closing the 'A' PORV is plausible as a reduction in Master pressure controller setpoint lowers the 'A' PORV setpoint and it could have opened the PORV. However, for the given pressures, the difference in setpoint and actual pressure is $2235 - 2160 = 75$ psig and the PORVs would not have opened.</p> <p>D is incorrect but plausible. Operation of the DECREASE pushbutton is the correct action and it will cause the spray valves to close. Closing the 'A' PORV is plausible as a reduction in setpoint lowers the 'A' PORV setpoint which is controlled by the Master pressure controller and it could have opened the PORV. However, for the given pressures, the difference in setpoint and actual pressure is $2235 - 2160 = 75$ psig and the PORVs would not have opened.</p>		
Technical Reference(s):	OS1201.06, "PZR Pressure Instrument/Component Failure"	
Proposed references to be provided to applicants during examination:	None	

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Learning Objective:	SBK LOP L8027I 05		
Question Source:	Bank #	X	TEB 28613
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:		2013 Seabrook NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	(10)	
	55.43		
Comments: Question #8 from 2018 Seabrook NRC Exam is a modified version of this question.			

B is correct. The required cooldown temperature per E-3 step 7 is 495 °F. The given ruptured SG pressure is between the 1100 and 1150 psig. Step 7a directs the required core exit temperature to be based upon pressure equal to or less than the lowest ruptured SG pressure. For this reason, the required temperature is 495 °F, not 500 °F. With a loss of offsite power, the condenser steam dumps are not available because no circulating water pumps are running. The ASDVs must be used.

A is incorrect but plausible. Required cooldown temperature is correct. Steam dumps are the preferred method of performing the cooldown, however steam dumps are not available with the loss of offsite power.

C is correct but plausible. If the student is unable to correctly apply step 7a to choose the required temperature based upon the pressure equal to or less than but instead uses the higher pressure, this answer could be chosen. Steam dumps are unavailable with the loss of offsite power.

D is incorrect but plausible. If the student is unable to correctly apply step 7a to choose the required temperature based upon the pressure equal to or less than but instead uses the higher pressure, this answer could be chosen. The ASDVs will be used with the loss of offsite power.

Technical Reference(s):		E-3, "Steam Generator Tube Rupture" Rev 45		
Proposed references to be provided to applicants during examination:				E-3, step 7a (page 7 only)
Learning Objective:	SBK LOP L1205I 02			
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	x		
Question History:				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41	(7)		

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	55.43	
Comments:		

Examination Outline Cross-reference:		Level	RO	SRO
Q8		Tier #	1	
		Group #	1	
		K/A #	000054 (CE E06) Loss of Main Feedwater /4 AK1.01 Knowledge of the operational implications of the following concepts as they apply to: Loss of Main Feedwater (MFW): MFW line break depressurizes the S/G (similar to a steam line break)	
		Importance Rating	4.1	
Proposed Question:				
<p>What is the basis for isolating all <u>feedwater</u> to a faulted SG in E-2, "Faulted Steam Generator Isolation"?</p> <p>A. Minimizes the temperature increase inside containment. B. Maximizes cool down capability of intact SGs. C. Minimize containment flooding concerns. D. Maximizes RCS heatup.</p>				
Proposed Answer:	B.			
Explanation (Optional):				
<p>B is correct. Isolation of the feedwater to the faulted SG maximizes the cool down capability of the nonfaulted loops following a feedline break.</p> <p>A is incorrect but plausible. Isolating feedwater flow to a faulted SG inside containment will mitigate the containment temperature increase but this is not the basis of the action.</p>				

<p>C is incorrect but plausible. Isolating emergency feedwater flow will reduce the volume of water added to containment, though this is not the basis.</p> <p>D is incorrect but plausible. Isolating feedwater flow to a faulted SG will cause an RCS heatup once the SG is dry. This is not the basis of isolating flow however.</p>			
<p>Technical Reference(s):</p>		<p>Basis document for E-2, Rev 3 page 35.</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1207I 02</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>x</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(8), (10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q9	Tier #		
	Group #		
	K/A #	000055 (EPE 55) Station Blackout / 6 EK1.02 Knowledge of the operational implications of the following concepts as they apply to the Station Blackout: Natural circulation cooling	
	Importance Rating	4.1	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A station blackout has occurred. • Subcooling is 35 °F and lowering. • SG Pressures are 1120 psig and stable. • RCS Hot Leg Temperatures are 620 °F and increasing. • CETCs are 625 °F and increasing. • RCS Cold Leg Temperatures are 560 °F and stable. <p>What is the status of natural circulation and what actions are required?</p> <p>A. Natural Circulation <u>IS</u> established. Throttle closed ASDVs to conserve SG inventory.</p> <p>B. Natural Circulation <u>IS</u> established. Throttle closed Condenser Steam Dumps to conserve SG inventory.</p> <p>C. Natural Circulation <u>IS NOT</u> established. Increase dumping steam from SGs with the ASDVs.</p> <p>D. A Natural Circulation <u>IS NOT</u> established. Increase dumping steam from SGs with the Condenser Steam Dumps.</p>			
Proposed Answer:	C.		

Explanation (Optional):			
<p>C is correct. The conditions listed indicate that natural circulation as defined in ECA-0.1 is not established. RCS subcooling is insufficient and RCS hot leg and CETC temperatures are increasing. The operator is required to increase dumping steam per step 14 RNO of ECA-0.1. Offsite power is not available and thus the condenser steam dumps are not available. The ASDVs must be used.</p> <p>A is incorrect but plausible. The conditions listed indicate that natural circulation as defined in ECA-0.1 is not established. RCS subcooling is insufficient and RCS hot leg and CETC temperatures are increasing. If the student incorrectly diagnoses the status of natural circulation, they may incorrectly apply step 17 of ECA-0.1 to stabilize plant conditions which includes stabilizing SG levels.</p> <p>B is incorrect but plausible. The conditions listed indicate that natural circulation as defined in ECA-0.1 is not established. RCS subcooling is insufficient and RCS hot leg and CETC temperatures are increasing. If the student incorrectly diagnoses the status of natural circulation, they may incorrectly apply step 17 of ECA-0.1 to stabilize plant conditions which includes stabilizing SG levels.</p> <p>D is incorrect but plausible. The conditions listed indicate that natural circulation as defined in ECA-0.1 is not established. RCS subcooling is insufficient and RCS hot leg and CETC temperatures are increasing. The operator is required to increase dumping steam per step 14 RNO of ECA-0.1. The condenser steam dumps are the preferred method of dumping steam in order to preserve secondary inventory, however they are unavailable due to the loss of offsite power. The ASDVs must be used.</p>			
Technical Reference(s):		ECA-0.1, "Loss of All AC Power Recovery Without SI Required".	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1210I 03		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	X	
Question History:			

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Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	(8), (10)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q10		Tier #	1	
		Group #	1	
		K/A #	000056 (APE 56) Loss of Offsite Power / 6 2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.	
		Importance Rating	3.8	
Proposed Question:				
<p>While performing a cool down in ECA-0.0, "Loss of All AC Power", what is the maximum allowed cool down rate and what is the purpose of performing this cooldown?</p> <p>A. Less than 100 °F/hr; to minimize RCS inventory loss through the RCP seals.</p> <p>B. The maximum rate achievable with the ASDVs; to minimize RCS inventory loss through the RCP seals.</p> <p>C. Less than 100 °F/hr; to establish conditions allowing for shutdown cooling with RHR once power is restored.</p> <p>D. The maximum rate achievable with the ASDVs; to establish conditions allowing for shutdown cooling with RHR once power is restored.</p>				
Proposed Answer:	A.			

Explanation (Optional):			
<p>A is correct. A note in ECA-0.0 directs the operators to perform the cool down at a rate near 100 °F/hr. The purpose of the cool down is to minimize the RCS inventory loss while cooling the RCP seals in a controlled manner.</p> <p>B is incorrect but plausible. A note in ECA-0.0 directs the operators to perform the cool down at a rate near 100 °F/hr. It is a common misconception that because of the potential consequences of a sustained loss of all AC power, a maximum cool down rate would be directed by the procedure. The purpose of the cool down is to minimize the RCS inventory loss while cooling the RCP seals in a controlled manner.</p> <p>C is incorrect but plausible. A note in ECA-0.0 directs the operators to perform the cool down at a rate near 100 °F/hr. It is a common misconception that the purpose of this cool down is to reduce temperature so that conditions can be established to utilize RHR to remove decay heat. At this point in the procedure the RHR pumps may have power but the cooldown will be performed on natural circulation.</p> <p>D is incorrect but plausible. A note in ECA-0.0 directs the operators to perform the cool down at a rate near 100 °F/hr. It is a common misconception that because of the potential consequences of a sustained loss of all AC power, a maximum cool down rate would be directed by the procedure. It is a common misconception that the purpose of this cool down is to reduce temperature so that conditions can be established to utilize RHR to remove decay heat. At this point in the procedure the RHR pumps may have power but the cooldown will be performed on natural circulation.</p>			
Technical Reference(s):		ECA-0.0, "Loss of All AC Power" Rev 55	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1210I 02		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	X	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		

2020 Seabrook Station NRC Written Exam
ES-401-5 Written Examination Question Worksheet

10 CFR Part 55 Content:	55.41	(10)
	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO										
Q11	Tier #	1											
	Group #	1											
	K/A #	000057 (APE 57) Loss of Vital AC Instrument Bus / 6 AK3.01 Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus.											
	Importance Rating	4.1											
Proposed Question:													
<p>While performing an EOP with EDE-PP-1A deenergized, what actions must be taken to start 'A' train ECCS equipment (1) and why (2)?</p> <table style="width: 100%; border: none;"> <tr> <td style="text-align: center; width: 50%;">(1)</td> <td style="text-align: center; width: 50%;">(2)</td> </tr> <tr> <td>A. SI must be manually actuated.</td> <td>The slave relays have lost power.</td> </tr> <tr> <td>B. SI must be manually actuated.</td> <td>The SSPS master relays have lost power.</td> </tr> <tr> <td>C. Equipment must be manually started.</td> <td>The slave relays have lost power.</td> </tr> <tr> <td>D. Equipment must be manually started.</td> <td>The SSPS master relays have lost power.</td> </tr> </table>				(1)	(2)	A. SI must be manually actuated.	The slave relays have lost power.	B. SI must be manually actuated.	The SSPS master relays have lost power.	C. Equipment must be manually started.	The slave relays have lost power.	D. Equipment must be manually started.	The SSPS master relays have lost power.
(1)	(2)												
A. SI must be manually actuated.	The slave relays have lost power.												
B. SI must be manually actuated.	The SSPS master relays have lost power.												
C. Equipment must be manually started.	The slave relays have lost power.												
D. Equipment must be manually started.	The SSPS master relays have lost power.												
Proposed Answer:	C.												
Explanation (Optional):													
<p>C is correct. Without power to EDE-PP-1A, the slave relays for all 'A' train ECCS equipment are deenergized and cannot be started via a manual or automatic SI actuation. The individual components must be started from the MCB.</p> <p>A is incorrect but plausible. It is a common misconception that a manual safety injection will successfully start components with slave relays deenergized.</p> <p>B is incorrect but plausible. It is a common misconception that a manual safety injection will successfully start components with slave relays deenergized. The SSPS master relays use</p>													

<p>redundant power supplies. In the case of 'A' train, the master relays are powered from PP-1A and PP-1C. The master relays are not deenergized with a loss of PP-1A only.</p> <p>D is incorrect but plausible. Individual components must be manually started but this is because the slave relays are deenergized, not the master relays.</p>			
<p>Technical Reference(s):</p>		<p>OS1247.01, "Loss of a Vital 120 VAC Instrument Panel" Rev 21</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L8056I 07</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>X</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(5), (10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:		Level	RO	SRO
Q12		Tier #	1	
		Group #	1	
		K/A #	000058 (APE 58) Loss of DC Power / 6 AA2.01 Ability to determine and interpret the following as they apply to the Loss of DC Power: That a loss of dc power has occurred; verification that substitute power sources have come on line.	
		Importance Rating	3.7	
Proposed Question:				
<p>Following maintenance activities on 125 V DC Bus 11B, the following alarms are observed:</p> <ul style="list-style-type: none"> • D6072, "Battery Charger 1B Output BKR Open" • D6633, "Batt 1B Discharging" <p>Which of the following alignments is consistent with these alarms?</p> <p>A. The portable battery charger supplying DC Bus 11B. B. DC Bus 11B on its alternate battery supply. C. Battery 1B in parallel with battery 1D. D. Battery 1B supplying DC Bus 11B.</p>				
Proposed Answer:		D.		
Explanation (Optional):				
<p>D is correct. With DC bus 11B supplied by battery 1B, the given alarms will occur. This is not true for any of the other possible answers.</p> <p>A is incorrect but plausible. It was misunderstood during the events leading to this LER that</p>				

<p>breaker DN4 is down stream of the connection point for both the normal and portable battery chargers. If the portable charger were in service, this alarm would not be in.</p> <p>B is incorrect but plausible. Placing the DC bus on an alternate battery supply is commonly done in order to support charging of the respective battery. However, this would not result in D6072 alarm.</p> <p>C is incorrect but plausible. It is a common misconception that charging of one battery results in parallel battery operation.</p>			
Technical Reference(s):		OS1248.01, "Loss of a 125 VDC Bus" Rev 13	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8017I 13		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	X	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
<p>Comments: New question developed based upon LER 443-06004. During this event, TS 3.8.2.1 was violated as a result of DC Bus 11B being left supplied by battery 1B only. This occurred following maintenance activities. Licensed operators initially failed to identify the significance of the control room alarms received and that they were not expected alarms. The crew was unable to verify that the substitute power source had come on-line, i.e. the vital battery is now powering the DC bus.</p>			

Examination Outline Cross-reference:		Level	RO	SRO
Q13		Tier #	1	
		Group #	1	
		K/A #	000062 (APE 62) Loss of Nuclear Service Water / 4 AA2.03 Ability to determine and interpret the following as they apply to the Loss of Nuclear Service Water: The valve lineups necessary to restart the SWS while bypassing the portion of the system causing the abnormal condition.	
		Importance Rating	2.6	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • A large Nor'easter has caused debris to be carried over into the SW fore bay. • A valid TA occurs in both trains. • While the crew is processing step 6h of OS1216.01, "Degraded Ultimate Heat Sink" to check SW strainer D/P, the NSO reports that the train 'A' SW strainer D/P is 15 psig and rising. <p>What actions are necessary in response to these conditions?</p> <p>A. Bypass the 'A' SW strainer. B. Manually wash all SW screens. C. Swap 'A' train of SW back to ocean. D. Throttle SW flow from the PCCW heat exchanger.</p>				
Proposed Answer:		A.		
Explanation (Optional):				

<p>A is correct. Per OS1216.01 step 6h if SW strainer D/P is greater than 10 psid the bypass valve for the strainer will be opened.</p> <p>B is incorrect but plausible. SW screens will be washed in response to high screen D/P not high strainer D/P.</p> <p>C is incorrect but plausible. If a running SW cooling tower pump discharge pressure is degraded and the ocean SW loop is available, the SW loop will be realigned back to the ocean per step 8 RNO of SO1216.01.</p> <p>D is incorrect but plausible. If adequate SW flow cannot be established to the SCCW system, the PCCW and DG loads will be throttled to increase system pressure.</p>			
<p>Technical Reference(s):</p>		<p>OS1216.01, "Degraded Ultimate Heat Sink" Rev 23.</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1193I 02</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>X</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(7)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO																				
Q14	Tier #	1																					
	Group #	1																					
	K/A #	000065 (APE 65) Loss of Instrument Air / 8 AK3.08 Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: Actions contained in EOP for loss of instrument air.																					
	Importance Rating	3.7																					
Proposed Question:																							
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • The service air (SA) 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-8005 now indicate 98 psig and increasing. <p>How will the SA header be returned to service (1) and what is the reason for this method (2)?</p> <table border="0" style="width: 100%;"> <thead> <tr> <th style="width: 50%;"></th> <th style="width: 50%; text-align: center;">(1)</th> <th style="width: 50%;"></th> <th style="width: 50%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A. Cycle SA-V-92/93 MCB control switch from open to close.</td> <td></td> <td>SA header must be slowly pressurized to avoid low pressure isolation.</td> <td></td> </tr> <tr> <td>B. Cycle SA-V-92/93 MCB control switch from open to close.</td> <td></td> <td>SA header must be slowly pressurized to avoid loss of plant control.</td> <td></td> </tr> <tr> <td>C. Hold MCB switch for SA-V-92/93 in the open position.</td> <td></td> <td>SA header must be slowly pressurized to avoid low pressure isolation.</td> <td></td> </tr> <tr> <td>D. Hold MCB switch for SA-V-92/93 in the open position.</td> <td></td> <td>SA header must be slowly pressurized to avoid loss of plant control.</td> <td></td> </tr> </tbody> </table>					(1)		(2)	A. Cycle SA-V-92/93 MCB control switch from open to close.		SA header must be slowly pressurized to avoid low pressure isolation.		B. Cycle SA-V-92/93 MCB control switch from open to close.		SA header must be slowly pressurized to avoid loss of plant control.		C. Hold MCB switch for SA-V-92/93 in the open position.		SA header must be slowly pressurized to avoid low pressure isolation.		D. Hold MCB switch for SA-V-92/93 in the open position.		SA header must be slowly pressurized to avoid loss of plant control.	
	(1)		(2)																				
A. Cycle SA-V-92/93 MCB control switch from open to close.		SA header must be slowly pressurized to avoid low pressure isolation.																					
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D. Hold MCB switch for SA-V-92/93 in the open position.		SA header must be slowly pressurized to avoid loss of plant control.																					

Proposed Answer:				A.	
Explanation (Optional):					
<p>A is correct. Step 9 of OS1242.01 directs to maintain IA dryer outlet >95 psig and cycle open SA-V-92/93 to re pressurize 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, which would cause an isolation. This is emphasized in a caution stating “The service air header must be pressurized slowly to prevent a service air header low pressure isolation at 90 PSIG”.</p> <p>B is incorrect but plausible. The highest priority for a loss of instrument air is to restore air to avoid a loss of plant control due to essential valves failing closed, e.g. PCCW containment isolations and main feedwater regulating valves. It is plausible that this is the basis for the actions required to restore SA air. However, low pressure isolation will avoid this loss of plant control if the header is restored incorrectly.</p> <p>C is incorrect but plausible. OS1242.01 contains a note stating, “the service air header can be pressurized by holding the switch for SA V92/SA V93 open”. However, the SA header must be pressurized slowly by cycling the control switch. Part (2) is correct.</p> <p>D is incorrect but plausible. OS1242.01 contains a note stating, “the service air header can be pressurized by holding the switch for SA V92/SA V93 open”. However, the SA header must be pressurized slowly by cycling the control switch. The reason for the strategy of cycling the open switch is to prevent a low pressure isolation, not avoid a loss of plant control.</p>					
Technical Reference(s):			ON1242.01, “Loss of Instrument Air”.		
Proposed references to be provided to applicants during examination:					None
Learning Objective:	SBK LOP L1194I 02				
Question Source:	Bank #				
	Modified Bank#				(Note changes or attach Parent)
	New	x			
Question History:					
Question Cognitive	Memory or Fundamental Knowledge				

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Level:	Comprehension or Analysis		X	
10 CFR Part 55 Content:	55.41	(5), (10)		
	55.43			
Comments:				

Examination Outline Cross-reference:		Level	RO	SRO
Q15		Tier #	1	
		Group #	1	
		K/A #	000077 (APE 77) Generator Voltage and Electric Grid Disturbances / 6 AA1.03 Ability to operate and/or monitor the following as they apply to Generator Voltage and Electric Grid Disturbances: Voltage regulator controls.	
		Importance Rating	3.8	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> The main generator is paralleled to the grid with the voltage regulator in AUTOMATIC sending 100 MVAR out. <p>Which of the following will occur if the operator places the Voltage Adjust switch from normal to lower?</p> <p>A. MWs decrease. B. MVARs decrease. C. Power factor decreases. D. Apparent power remains constant.</p>				
Proposed Answer:	B.			
<p>Explanation (Optional):</p> <p>B is correct. Lowering main generator voltage in this condition results in a decrease in reactive load.</p> <p>A is incorrect but plausible. It is a common misconception that lowering generator voltage will result in a decrease in MW loading.</p> <p>C is incorrect but plausible. For the conditions given lowering voltage will result in lowering</p>				

reactive load and an increase in the power factor.			
D is incorrect but plausible. This incorrect because as reactive load is decreased, apparent power lowers. Apparent power and true power are routinely confused and for the operator to correctly manipulate main control board controls, this difference must be understood.			
Technical Reference(s):		ON1000.10, "Operation at Power" Figure 12.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8016I 07		
Question Source:	Bank #	X	10111
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2009 Comanche Peak NRC Exam (same K/A)		
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5), (10)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q16	Tier #	1	
	Group #	1	
	K/A #	(W E04) LOCA Outside Containment / 3 EK2.2 Knowledge of the interrelations between the (LOCA Outside Containment) and the following: Facility's heat removal systems, including primary coolant, emergency coolant, the decay heat removal systems, and relations between the proper operation of these systems to the operation of the facility.	
	Importance Rating	3.8	
Proposed Question:			
<p>The crew has entered ECA-1.2, "LOCA OUTSIDE CONTAINMENT".</p> <p>After closing RH-V-14, "RHR Train A discharge to the RCS" and RH-V-22, "RHR Train A cross-connect" and placing the 'A' train RHR and CBS pumps in pull-to-lock, the following conditions exist:</p> <ul style="list-style-type: none"> • ECCS flow is decreasing • RCS pressure is 1100 psig and slowly increasing <p>Which of the following indicates the status of the LOCA and the FIRST procedure transition that will be made, if any?</p> <p>A. The LOCA <u>is isolated</u>. The crew will transition to E-0, "Reactor Trip or Safety Injection", step 1.</p> <p>B. The LOCA <u>is not isolated</u>. The crew will continue with actions in ECA-1.2, "LOCA OUTSIDE CONTAINMENT".</p> <p>C. The LOCA <u>is isolated</u>. The crew will transition to E-1, "LOSS OF REACTOR OR SECONDARY COOLANT", step 1.</p> <p>D. The LOCA <u>is not isolated</u>. The crew will transition to ECA-1.1, "LOSS OF EMERGENCY COOLANT RECIRCULATION", step 1.</p>			

Proposed Answer:	C.	
Explanation (Optional):		
<p>C is correct. Per ECA-1.2, step #4 if RCS pressure is increasing due to successful leak isolation the crew will transition to E-1.</p> <p>A is incorrect but plausible. If the LOCA is outside containment only and not associated with another accident, ECA-1.2 may be entered directly from E-0 making it plausible that if the leak is isolated transition back to E-0 be required.</p> <p>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. However, the conditions as given indicate that the LOCA has been isolated.</p> <p>D is incorrect but plausible. The crew would transition to ECA-1.1 at step #4 of ECA-1.2 if RCS pressure continues to decrease due to the leak.</p>		
Technical Reference(s):	ECA-1.2, "LOCA Outside Containment".	
Proposed references to be provided to applicants during examination:	None	
Learning Objective:	SBK LOP L1209I 04	
Question Source:	Bank #	X TEB 29959
	Modified Bank#	(Note changes or attach Parent)
	New	
Question History:	2007 Seabrook NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	(7)
	55.43	
Comments:		

Examination Outline Cross-reference:		Level	RO	SRO
Q17		Tier #	1	
		Group #	1	
		K/A #	(W E05) Loss of Secondary Heat Sink / 4 EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	
		Importance Rating	3.7	
Proposed Question:				
<p>What is the basis for stopping all RCPs in FR-H.1, "Loss of Secondary Heat Sink"?</p> <p>A. It prevents core uncover if feedwater cannot be established</p> <p>B. It increases the time allowed to establish a higher flow rate for high pressure injection thus raising the cooldown rate.</p> <p>C. It extends the time to restore feed flow to the SGs by reducing RCS heat input, extending the effectiveness of the remaining water in the SGs.</p> <p>D. It allows for time to depressurize the intact SGs in order to reduce RCS pressure and inject accumulators.</p>				
Proposed Answer:	C.			
Explanation (Optional):				
<p>Question meets K/A by testing the student on FR-H.1 and the reason needed to manually operate the main control board switches for the RCPs.</p> <p>C is correct. Per FR-H.1 background document (step 4) the purpose of stopping all RCPs in FR-H.1 is to "extend the time to restore feed flow to the SGs".</p> <p>A is incorrect but plausible. Stopping the RCPs will extend time to core uncover in a small break LOCA making this plausible. It will not prevent core uncover in this case.</p> <p>B is incorrect but plausible. High pressure injection will be established by actuating SI when</p>				

required in FR-H.1, however stopping the RCPs is not related to this.			
D is incorrect but plausible. Depressurization of the SGs to allow for accumulator injection is a strategy in FR-C.1 which is loosely related to FR-H.1.			
Technical Reference(s):		Background document for FR-H.1 Rev 3	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1211I 01		
Question Source:	Bank #	X	17769
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2013 Byron NRC Exam (same K/A)		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q18	Tier #	1	
	Group #	1	
	K/A #	(W E11) Loss of Emergency Coolant Recirculation / 4 EK3.3 Knowledge of the reasons for the following responses as they apply to the (Loss of Emergency Coolant Recirculation): Manipulation of controls required to obtain desired operating results during abnormal, and emergency situations.	
	Importance Rating	3.8	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Large Break LOCA. • 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": <ul style="list-style-type: none"> ➤ RH-P-8A is lost due to a sheared 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: <ul style="list-style-type: none"> ➤ "If suction source is lost to any ECCS or spray pump, the pump should be stopped". <p>What pumps should be stopped and why?</p> <p>A. Both SI pumps, only. The low RWST level could cause cavitation. B. Both Charging pumps, only. The low RWST level could cause cavitation. C. Both Charging pumps <u>and</u> both SI pumps, only. The pumps were being supplied suction from</p>			

the RHR pumps. D. Both CBS pumps, both SI pumps <u>and</u> both Charging pumps. The pumps were being supplied suction from the RHR pumps.				
Proposed Answer:	C.			
Explanation (Optional):				
<p>C is correct. With cold leg recirculation established, the RHR pumps are supplying the suction of the charging and SI pumps. Loss of the RHR pumps will result in a loss of suction source for these pumps. In accordance with the given caution, both SI and both charging pumps must be stopped. The CBS pumps are taking suction from the containment recirculation sumps and are not impacted by the loss of the RHR pumps.</p> <p>A is incorrect but plausible. Actions in ECA-1.1 will reduce the number of running ECS pumps to the minimum required. If the student does not understand the given note, this is a plausible answer based upon these actions and that the RWST level is low at 100,000 gallons. ECA-1.1 will stop pumps if RWST level is less than 80,000 gallons.</p> <p>B is incorrect but plausible. Actions in ECA-1.1 will reduce the number of running ECS pumps to the minimum required. If the student does not understand the given note, this is a plausible answer based upon these actions. ECA-1.1 also requires that the charging pumps be secured if RWST level is < 40,000 gallons. This additionally discriminates from students understanding the caution and those that do not.</p> <p>D is incorrect but plausible. If the student does not understand the given caution and the reasons for the loss of suction to the running pumps it is conceivable that in this procedure the CBS pumps be secured as well. ECA-1.1 will secure the CBS pumps if swap over to the containment sumps cannot be achieved.</p>				
Technical Reference(s):	ECA-1.1, "Loss of Emergency Coolant Recirculation" Rev 38			
Proposed references to be provided to applicants during examination:	None			
Learning Objective:	SBK LOP L1209I 03			
Question Source:	Bank #	x	TEB 22246	
	Modified Bank#			(Note changes or attach Parent)
	New			

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Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(5), (10)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO										
Q19		Tier #	1											
		Group #	2											
		K/A #	000001 (APE 1) Continuous Rod Withdrawal / 1 AA1.01 Ability to operate and / or monitor the following as they apply to the Continuous Rod Withdrawal: Bank select switch											
		Importance Rating	3.5											
Proposed Question:														
<p>Plant conditions:</p> <ul style="list-style-type: none"> 75% power and stable following a down power. Rod control is in AUTO. Control bank 'D' begins to withdraw. <p>In accordance with OS1202.04, "Continuous Control Rod Withdrawal" the initial action is to place the rod bank selector switch in <u> (1) </u> and if that action fails to stop rod motion, the crew is required to <u> (2) </u>.</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. MANUAL</td> <td>trip the reactor and go to E-0</td> </tr> <tr> <td>B. MANUAL</td> <td>attempt to insert CBD by placing the In/Hold/Out switch to 'In'</td> </tr> <tr> <td>C. CBD</td> <td>trip the reactor and go to E-0</td> </tr> <tr> <td>D. CBD</td> <td>attempt to insert CBD by placing the In/Hold/Out switch to 'In'</td> </tr> </table>					(1)	(2)	A. MANUAL	trip the reactor and go to E-0	B. MANUAL	attempt to insert CBD by placing the In/Hold/Out switch to 'In'	C. CBD	trip the reactor and go to E-0	D. CBD	attempt to insert CBD by placing the In/Hold/Out switch to 'In'
(1)	(2)													
A. MANUAL	trip the reactor and go to E-0													
B. MANUAL	attempt to insert CBD by placing the In/Hold/Out switch to 'In'													
C. CBD	trip the reactor and go to E-0													
D. CBD	attempt to insert CBD by placing the In/Hold/Out switch to 'In'													
Proposed Answer:	A.													
Explanation (Optional):														
<p>A is correct. OS1210.04, "Continuous Control Rod Withdrawal" directs the crew to place the rod bank selector switch in manual and if rod motion continues, to trip the reactor and go to E-0.</p> <p>B is incorrect but plausible. OS1210.04, "Continuous Control Rod Withdrawal" directs the crew to place the rod bank selector switch in manual. Rod insertion is required in FR-S.1 if the reactor fails to trip. It is not an action in OS1202.04.</p>														

<p>C is incorrect but plausible. The stem of the question gives that it is control bank D that is withdrawing making placing the switch in CBD plausible. If rod motion continues the crew is required to trip the reactor and go to E-0.</p> <p>D is incorrect but plausible. The stem of the question gives that it is control bank D that is withdrawing making placing the switch in CBD plausible. Rod insertion is required in FR-S.1 if the reactor fails to trip. It is not an action in OS1202.04.</p>			
<p>Technical Reference(s):</p>		<p>OS1210.04, "Continuous Control Rod Withdrawal"</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1184I 12</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>x</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(7)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:		Level	RO	SRO
Q20		Tier #	1	
		Group #	2	
		K/A #	000036 (APE 36; BW/A08) Fuel Handling Incidents / 8 AK3.02 Knowledge of the reasons for the following responses as they apply to the Fuel Handling Incidents: Interlocks associated with fuel handling equipment	
		Importance Rating	2.9	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Mode 6 with refueling operations in progress. • An assembly is being lifted out of the core. <p>If the fuel assembly binds against another adjacent assembly, upward motion of the hoist will be automatically stopped to prevent fuel assembly damage.</p> <p>What refueling machine interlock provides this protection?</p> <p>A. Hoist encoder error interlock. B. Load comparison error. C. Hoist raise interlock. D. Hoist over load.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
<p>D is correct. Hoist over load is activated when load cell average weight is above the reference weight.</p> <p>A is incorrect but plausible. Hoist encoder error interlock is activated when the two mast position</p>				

encoders differ by a set amount in either direction.				
B is incorrect but plausible. Load comparison error is activated when the two load cells differ by a set amount.				
C is incorrect but plausible. Hoist raise interlock prevents raising the hoist with a load and an unlatched gripper.				
Technical Reference(s):		OS1015.04, "Refueling Machine Operation".		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8060I 05			
Question Source:	Bank #			
	Modified Bank#	x	11122	(Note changes or attach Parent)
	New			
Question History:	2010 Beaver Valley 2 NRC Exam			
Question Cognitive Level:	Memory or Fundamental Knowledge		x	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	(5), (10)		

	55.43	
<p>Comments: Original question</p> <p>The unit is in Mode 6. A fuel assembly is being lowered into the core.</p> <p>IF the fuel assembly BINDS against another fuel assembly, downward motion of the hoist will be automatically stopped to prevent fuel assembly damage.</p> <p>What manipulator crane interlock provides this protection?</p> <ul style="list-style-type: none">A. OverloadB. UnderloadC. Tube DownD. Bridge-Trolley Hoist <p>Answer B.</p>		

Examination Outline Cross-reference:		Level	RO	SRO
Q21		Tier #	1	
		Group #	2	
		K/A #	000037 (APE 37) Steam Generator Tube Leak / 3 AK1.02 Knowledge of the operational implications of the following concepts as they apply to Steam Generator Tube Leak: Leak rate vs. pressure drop.	
		Importance Rating	3.5	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A steam generator tube leak has occurred. • The crew is implementing OS1227.02, "Steam Generator Tube Leak". • The affected steam generator has been isolated and the crew is in the process of performing the RCS cooldown. <p>Which of the following actions will ensure that the cooldown of the RCS does not depressurize the affected steam generator?</p> <p>A. Stopping the reactor coolant pump in the affected loop.</p> <p>B. Maintaining RCS pressure above the affected ASDV setpoint.</p> <p>C. Raising the affected steam generator's ASDV setpoint to 1185 psig.</p> <p>D. Maintaining the affected steam generator's water level above the top of the u-tubes.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
D is correct. Maintaining the affected steam generators water level above the u-tubes will maintain thermal partitioning between the RCS and steam generator such that the generator does not depressurize, which would result in further ΔP driving force for the tube leak.				

<p>A is incorrect but plausible. Stopping the RCP could conceivably result in removal of forced flow of cooler RCS water to the affected steam generator however, this is not one of the procedural strategies.</p> <p>B is incorrect but plausible. Maintaining RCS pressure above the ASDV setpoint is conceptually tied to the theoretical relationship between leak rate and ΔP however, raising RCS pressure would have the effect if increasing ΔP.</p> <p>C is incorrect but plausible. Raising the SG ASDV setpoint would allow SG pressure to drift higher if the RCS were adding heat to the steam generator. The increase in steam generator pressure is conceptually tied to the theoretical relationship between leak rate and ΔP however, this is not one of the procedural strategies.</p>			
<p>Technical Reference(s):</p>		<p>OS1227.02, "Steam Generator Tube Leak" Rev 20</p> <p>ARG-3, Background Document for Steam Generator Tube Leak AOP.</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1190I 04</p>		
<p>Question Source:</p>	<p>Bank #</p>	<p>X</p>	<p>TEB 25538</p>
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>		
<p>Question History:</p>	<p>2018 Seabrook NRC Exam</p> <p>(Question used on one of the two previous NRC exams) Same K/A for EPE 38</p>		
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>X</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(8), (10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q22	Tier #	1	
	Group #	2	
	K/A #	000051 (APE 51) Loss of Condenser Vacuum / 4 AA2.02 Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: Conditions requiring reactor and/or turbine trip.	
	Importance Rating	3.9	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The crew is implementing ON1233.01, "Loss of Condenser Vacuum". • Generator output has been lowered to 330 MW electric. • Condenser vacuum is 24.8 "HgV and degrading. <p>What is the next required action?</p> <p>A. Continue to lower generator output until vacuum improves. B. Shift mechanical vacuum pump discharge. C. Trip the reactor. D. Trip the turbine.</p>			
Proposed Answer:	C.		
Explanation (Optional):			
<p>C is correct. Step 3 of ON1233.01 has the crew decrease power to restore vacuum. If load is decreased below 360 MWeI and vacuum is less than 25 "HgV a reactor trip is required.</p> <p>A is incorrect but plausible. The steps to decrease plant power to restore vacuum require that if power is decreased to less than 360 MWeI and vacuum continues to degrade, the reactor be tripped. Power is not decreased continuously until vacuum improves.</p> <p>B is incorrect but plausible. Step 3d of ON1233.01 has the operator shift the mechanical vacuum</p>			

<p>pump discharge to the atmosphere, but only after load is reduced and condenser vacuum remains above 25 "HgV.</p> <p>D is incorrect but plausible. 330 MWel corresponds to 25% power. Based upon this value the student could interpret that being below P-9 (45%) requires a turbine trip vs a reactor trip.</p>			
<p>Technical Reference(s):</p>		<p>ON1233.01, "Loss of Condenser Vacuum".</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L188I 08</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>X</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		
	<p>Comprehension or Analysis</p>		<p>X</p>
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q23	Tier #	1	
	Group #	2	
	K/A #	000032 (APE 32) Loss of Source Range Nuclear Instrumentation / 7 AA2.04 Ability to determine and interpret the following as they apply to the Loss of Source Range Nuclear Instrumentation: Satisfactory source-range/intermediate-range overlap.	
	Importance Rating	3.1	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The crew is performing a reactor start up with control rods. • Source range NI-31 and 32 are reading 5E4 CPS. • Intermediate range NI-35 and 36 are reading 2E-10 amps. • Startup rate is 0.2 DPM <p>NI-32 fails to 0 CPS.</p> <p>What action will the crew take in accordance with OS1000.07, "Approach to Criticality"?</p> <p>A. Initiate boration at 30 gpm. B. Fully insert all control banks. C. Verify proper overlap and block SR trips. D. Insert control banks to 10 steps above the RIL.</p>			

Proposed Answer:		C.	
Explanation (Optional):			
<p>C is correct. For the given plant conditions, P-6 is actuated (1/2 IR >1E-10). The crew will verify proper overlap between the source and intermediate range and block the SR trips, which will also de energize the SR detectors. The failure of the one SR detector does not prevent this action physically or administratively.</p> <p>B is incorrect but plausible. The startup using control rods would be performed with OS1000.07, "Approach to Criticality". Step 4.5 contains several unexpected conditions for which the required action is to fully insert all control banks. Loss of a single SR under the given conditions is not one of these conditions.</p> <p>A is incorrect but plausible. A loss of shutdown margin requires a boration of 30 gpm. This is not applicable to the given situation.</p> <p>D is incorrect but plausible. The startup using control rods would be performed with OS1000.07, "Approach to Criticality". Step 4.5 contains several unexpected conditions for which the required action is to fully insert all control banks. There are also conditions that require rods to be inserted to 10 steps above the RIL making this plausible.</p>			
Technical Reference(s):		OS1000.07, "Approach to Criticality" Rev 16	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8030I 03		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	X	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41	(10)	

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	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q24	Tier #	1	
	Group #	2	
	K/A #	000076 (APE 76) High Reactor Coolant Activity / 9 AK2.01 Knowledge of the interrelations between the High Reactor Coolant Activity and the following: Process radiation monitors.	
	Importance Rating	2.6	
Proposed Question:			
<p>At 100% power which of the following radiation monitors will provide the first direct indication of high RCS activity?</p> <p>A. RM6505-1 Condenser Air Evacuation. B. RM6520-1 Letdown Rad Monitor. C. RM6576-1 Post LOCA. D. RM6548-1 Alt Gas.</p>			
Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. The letdown rad monitor is used to diagnose high RCS activity.</p> <p>A is incorrect but plausible. The condenser air evacuation rad monitor would see radiation from high RCS activity but only if the SG U-tubes were not intact.</p> <p>C is incorrect but plausible. The post LOCA rad monitors are intended for post LOCA conditions but not primarily used to indicate high RCS activity.</p> <p>D is incorrect but plausible. The Alt Gas rad monitor is one of the primary means of diagnosing a leak in containment but not of diagnosing high RCS activity.</p>			

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Technical Reference(s):	N/A		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1181I 08		
Question Source:	Bank #	x	10633
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2009 Wolf Creek NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q25	Tier #	1	
	Group #	2	
	K/A #	(BW E08; W E03) LOCA Cooldown—Depressurization / 4 EK2.1 Knowledge of the interrelations between the (LOCA Cooldown and Depressurization) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	
	Importance Rating	3.6	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • ES-1.2, “Post LOCA Cooldown and Depressurization” is in progress. • ‘C’ RCP is running. • Both RHR pumps are stopped and in standby. • All RCS hot leg temperatures: 345 to 350 °F. • Core Exit Thermocouples: 360 °F • Wide Range RCS pressure: 325 psig • Pressurizer level: 45% slowly rising. • Two charging pumps are running in the ECCS injection mode. • The crew is performing step 13 to “Check If One Charging Pump Should Be Stopped”. • The required subcooling in step 13c is 74 °F. <p>Can one CCP be stopped at this time and why, or why not?</p> <p>A. Yes. An RHR pump must be started first. Adequate subcooling will be maintained after one CCP is stopped.</p>			

<p>B. Yes. Subcooling requirements do not apply at this temperature. An RHR pump does NOT need to be started. Adequate subcooling will be maintained after one CCP is stopped.</p> <p>C. No. Subcooling must be greater than the value required in step 13c.</p> <p>D. No. Subcooling requirements apply until hot leg temperatures are below 250 °F.</p>			
Proposed Answer:	A.		
Explanation (Optional):			
<p>A is correct. Per ES-1.2 step 13 if subcooling is below required values, an RHR pump must be started to ensure adequate subcooling remains once the CCP is stopped. Based on given conditions subcooling is: 325 psig = 340 psia, T_{sat} for 340 psia = 429 °F, 429-360=69 °F which is less than the given required subcooling in step 13c.</p> <p>B is incorrect but plausible. This is only true if an RHR pump is started prior to stopping one CCP.</p> <p>C is incorrect but plausible. The subcooling requirements apply unless RCS temperature is less than 360 °F and an RHR pump has been started.</p> <p>D is incorrect but plausible. 250 °F is an important value related to RHR operation in shutdown cooling mode. It is not a value considered in ES-1.2.</p>			
Technical Reference(s):	ES-1.2, "Post LOCA Cooldown and Depressurization" Background document ES-1.2 Rev3		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1204I 03		
Question Source:	Bank #	x	TEB 31628
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(8), (10)	

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	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO										
Q26	Tier #	1											
	Group #	2											
	K/A #	(CE A11**; W E08) RCS Overcooling—Pressurized Thermal Shock / 4 EK3.1 Knowledge of the reasons for the following responses as they apply to the (Pressurized Thermal Shock): Facility operating characteristics during transient conditions, including coolant chemistry and the effects of temperature, pressure, and reactivity changes and operating limitations and reasons for these operating characteristics.											
	Importance Rating	3.4											
Proposed Question:													
<p>While performing FR-P.1, “Response to Imminent Pressurized Thermal Shock Conditions”, the basis for terminating SI is ____ (1) ____ and if SI cannot be terminated it is desirable to start an RCP in order to ____ (2) ____.</p> <table border="0" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="width: 50%; text-align: center;">(1)</th> <th style="width: 50%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure</td> <td>mix the cold incoming SI water and the warm reactor coolant water</td> </tr> <tr> <td>B. the temperature soak requires SI to be secured</td> <td>mix the cold incoming SI water and the warm reactor coolant water</td> </tr> <tr> <td>C. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure</td> <td>prevent upper head voiding during depressurization</td> </tr> <tr> <td>D. the temperature soak requires SI to be secured</td> <td>prevent upper head voiding during depressurization</td> </tr> </tbody> </table>				(1)	(2)	A. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure	mix the cold incoming SI water and the warm reactor coolant water	B. the temperature soak requires SI to be secured	mix the cold incoming SI water and the warm reactor coolant water	C. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure	prevent upper head voiding during depressurization	D. the temperature soak requires SI to be secured	prevent upper head voiding during depressurization
(1)	(2)												
A. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure	mix the cold incoming SI water and the warm reactor coolant water												
B. the temperature soak requires SI to be secured	mix the cold incoming SI water and the warm reactor coolant water												
C. SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure	prevent upper head voiding during depressurization												
D. the temperature soak requires SI to be secured	prevent upper head voiding during depressurization												

Proposed Answer:				A.			
Explanation (Optional):							
<p>A is correct. Per the basis document for FR-P.1 Rev 3 page 25, the basis for terminating SI in this procedure is that SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure. If SI cannot be terminated RCPs will be started once conditions can be established. RCPs are started in order to mix the cold incoming SI water and the warm reactor coolant water and thereby decrease the likelihood of a PTS condition.</p> <p>B is incorrect but plausible. Step 24 of FR-P.1 will evaluate if an RCS soak is required. It is reasonable to assume that the SI flow will interfere with this soak and that is the basis for securing the SI. If SI cannot be terminated RCPs will be started once conditions can be established. RCPs are started in order to mix the cold incoming SI water and the warm reactor coolant water and thereby decrease the likelihood of a PTS condition.</p> <p>C is incorrect but plausible. Per the basis document for FR-P.1 Rev 3 page 25, the basis for terminating SI in this procedure is that SI flow may have contributed to the RCS cooldown or may prevent a subsequent reduction in RCS pressure. FR-P.1 contains a caution before step 16 that the upper head region may void during RCS depressurization if RCPs are not running making this plausible. There are however, no actions taken to start an RCP to prevent this from occurring.</p> <p>D is incorrect but plausible. Step 24 of FR-P.1 will evaluate if an RCS soak is required. It is reasonable to assume that the SI flow will interfere with this soak and that is the basis for securing the SI. FR-P.1 contains a caution before step 16 that the upper head region may void during RCS depressurization if RCPs are not running making this plausible. There are however, no actions taken to start an RCP to prevent this from occurring.</p>							
Technical Reference(s):				FR-P.1, "Response to Imminent Pressurized Thermal Shock Conditions" Rev 34 Background Document for FR-P.1 Rev 3.			
Proposed references to be provided to applicants during examination:						None	
Learning Objective:		SBK LOP L1208I 04					
Question Source:		Bank #					
		Modified Bank#				(Note changes or attach Parent)	
		New		x			

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Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5), (10)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q27		Tier #	1	
		Group #	2	
		K/A #	(W E15) Containment Flooding / 5 2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.	
		Importance Rating	4.3	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% 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: <ul style="list-style-type: none"> ○ Reactor Makeup Water (RMW) ○ Primary Component Cooling Water (PCCW) ○ Fire Protection (FP) <p>How will the PSO check that the penetrations are isolated, and which penetration(s) needs to be isolated, if any?</p> <p>A. The containment isolation valves for all three systems can be checked on the UL panels. PCCW will need to be isolated.</p> <p>B. The containment isolation valves for all three systems can be checked on the UL panels. All three panels should already be isolated.</p> <p>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.</p> <p>D. The RMW and PCCW isolations can be checked on the UL panels. The FP path can only be verified locally. All three paths should already be isolated.</p>				
Proposed Answer:	D.			

Explanation (Optional):			
<p>D is correct. A large break LOCA will generate a 'T' and a 'P' signal. RMW-V-30 isolates on a 'T' signal and indicates full closed on UL-3. PCCW isolation valves to containment isolate on a 'P' signal and indicate on the UL panel. FP-V-592 is locked closed in modes 1, 2, 3, and 4 and has no remote indications of valve position.</p> <p>A is incorrect but plausible. A large break LOCA will generate a 'T' and a 'P' signal. RMW-V-30 isolates on a 'T' signal and indicates full closed on UL-3. PCCW isolation valves to containment isolate on a 'P' signal and indicate on the UL panel. FP-V-592 is locked closed in modes 1, 2, 3, and 4 and has no remote indications of valve position.</p> <p>B is incorrect but plausible. A large break LOCA will generate a 'T' and a 'P' signal. RMW-V-30 isolates on a 'T' signal and indicates full closed on UL-3. PCCW isolation valves to containment isolate on a 'P' signal and indicate on the UL panel. FP-V-592 is locked closed in modes 1, 2, 3, and 4 and has no remote indications of valve position.</p> <p>C is incorrect but plausible. A large break LOCA will generate a 'T' and a 'P' signal. RMW-V-30 isolates on a 'T' signal and indicates full closed on UL-3. PCCW isolation valves to containment isolate on a 'P' signal and indicate on the UL panel. FP-V-592 is locked closed in modes 1, 2, 3, and 4 and has no remote indications of valve position.</p>			
Technical Reference(s):		FR-Z.2, " Response to Containment Flooding", Rev 19	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1212I 01		
Question Source:	Bank #	x	TEB 26915
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2013 Seabrook NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(10)	

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	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q28	Tier #	2	
	Group #	1	
	K/A #	003 (SF4P RCP) Reactor Coolant Pump K2.01 Knowledge of bus power supplies to the following: RCPS	
	Importance Rating	3.1	
Proposed Question:			
<p>Bus voltage on 13.8 kV Bus 2 begins to steadily decrease.</p> <p>What component(s) will automatically trip <u>first</u>?</p> <p>A. 'B' CW pump. B. 'B' and 'D' RCPs. C. 'C' and 'D' RCPs. D. 'A' and 'C' CW pumps.</p>			
Proposed Answer:	C.		
Explanation (Optional):			
<p>C is correct. C and D RCPs are powered from 13.8 kV Bus 2. As bus voltage steadily decreases, the RCPs trip at 70% bus voltage after 1/3 second. There is a common misconception as to the power supply to the RCPs and CW pumps. The pumps are powered by 13.8 kV bus 1 and 2. The A and B RCPs and A and C CW pumps are powered from bus 1 and the C and D RCPs and B CW pumps are powered from bus 2. The RCPs have a 1/3 second time delay. The CW pumps are stripped from the bus after a 1.5 second time delay.</p> <p>A is incorrect but plausible. The B CW pump is powered from bus 2 and it does get stripped from the bus, but only after the C and D RCPs.</p> <p>B is incorrect but plausible. The power supplies for the RCPs are a common misconception.</p> <p>D is incorrect but plausible. The CW pumps do get stripped from the bus but only after the RCPs. The power supplies for the CW pumps are a common misconception.</p>			

Technical Reference(s):		N/A		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8012I 27			
Question Source:	Bank #			
	Modified Bank#	x	13107	(Note changes or attach Parent)
	New			
Question History:	2013 Seabrook NRC Exam			
Question Cognitive Level:	Memory or Fundamental Knowledge		x	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	(7)		
	55.43			
<p>Comments:</p> <p>Original question:</p> <p>Bus voltage on 13.8 kV Bus 1 begins to steadily decrease.</p> <p>What component(s) will automatically trip <u>first</u>?</p> <p>A. 'C' CW Pump. B. 'A' and 'B' RCPs. C. 'A' and 'C' RCPs. D. 'A' and 'B' CW pumps.</p> <p>Answer B</p>				

Examination Outline Cross-reference:		Level	RO	SRO
Q29		Tier #	2	
		Group #	1	
		K/A #	004 (SF1; SF2 CVCS) Chemical and Volume Control K4.03 Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Protection of ion exchangers (high letdown temperature will isolate ion exchangers)	
		Importance Rating	2.8	
Proposed Question:				
<p>What condition will <u>directly</u> cause CS-TCV-129, "Demin Divert Valve" to reposition to bypass?</p> <p>A. Letdown moderating heat exchanger outlet temp as seen on CS-TI-386 >130 °F.</p> <p>B. Regen heat exchanger outlet temp as seen on CS-TI-127 >395 °F.</p> <p>C. High letdown temp as seen on CS-TI-130 >134 °F.</p> <p>D. High VCT temp as seen on CS-TI-116 >120 °F.</p>				
Proposed Answer:	C.			
Explanation (Optional):				
<p>C is correct. As shown in VPRO for alarm D4695, letdown temperature greater than 134 °F will cause TCV-129 to bypass the letdown demineralizers.</p> <p>A is incorrect but plausible. The letdown moderating heat exchanger is part of the BTRS (Boron Thermal Regeneration System) subsystem of CVCS. It is used to remove boron from the RCS at EOL. If inlet temperature to the BTRS demins exceeds 157 °F, TCV-129 will reposition to bypass making the given condition plausible. Letdown moderating heat exchanger outlet temp >130 °F will cause alarm point D4623 only.</p>				

<p>B is incorrect but plausible. A high regen heat exchanger outlet temperature is indicative of high letdown flow or low charging flow. This indication is upstream of the letdown heat exchanger and is not the direct input to CS-TCV-129. The condition given will cause alarm D4675 to actuate only.</p> <p>D is incorrect but plausible. High VCT temperature is indicative of inadequate cooling in the letdown system. If temperature could not be recovered, the operator would isolate letdown manually. This is not a direct input to TCV-129.</p>			
Technical Reference(s):		VPRO D4695	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8024I05		
Question Source:	Bank #	<input type="checkbox"/>	<input type="checkbox"/>
	Modified Bank#	<input type="checkbox"/>	<input type="checkbox"/>
	New	<input checked="" type="checkbox"/>	<input type="checkbox"/>
Question History:	<input type="checkbox"/>		
Question Cognitive Level:	Memory or Fundamental Knowledge	<input checked="" type="checkbox"/>	<input type="checkbox"/>
	Comprehension or Analysis	<input type="checkbox"/>	<input type="checkbox"/>
10 CFR Part 55 Content:	55.41	(7)	
	55.43	<input type="checkbox"/>	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q30	Tier #	2	
	Group #	1	
	K/A #	004 (SF1; SF2 CVCS) Chemical and Volume Control K4.08 Knowledge of CVCS design feature(s) and/or interlock(s) which provide for the following: Hydrogen control in RCS	
	Importance Rating	2.8	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • The PSO performs a routine CVCS Volume Control Tank divert. • During the divert evolution the PSO notices that the 'A' Reactor Coolant Pump #1 seal return flow is 2.5 gpm and slowly <u>rising</u>. <p>Why is the 'A' RCP #1 seal return flow rising and what action should the operator take?</p> <p>A. VCT pressure has increased causing an increase in both seal injection and seal return flow. The operator should maintain VCT pressure less than 25 psig.</p> <p>B. VCT pressure has decreased causing a decrease in #1 seal return backpressure. The operator should maintain VCT hydrogen pressure greater than 15 psig.</p> <p>C. The VCT divert flow path branches off of the charging flow path causing a reduction in both seal injection and seal leak off flow. The operator should adjust CS-LK-185, VCT Divert Control to maintain adequate seal injection flow.</p> <p>D. The VCT divert flow path branches off of the seal return line causing a decrease in #1 seal return backpressure. The operator should adjust CS-LK-185, VCT Divert Control to maintain adequate seal return backpressure.</p>			
Proposed Answer:	B.		
Explanation (Optional):			
B is correct. The Reactor Coolant Pump #1 Seal Return line is routed to the bottom or outlet of the VCT. VCT pressure has a direct impact on seal return backpressure. When the VCT is diverted			

the tank inlet flow from letdown is re-routed. This causes a resulting drop in VCT pressure. The drop in VCT pressure results in a drop in seal return backpressure and an increase in seal return flow. The procedural guidance for performing a VCT divert (procedure OS1002.02 section 4.41) directs the operator to verify that VCT pressure is being maintained greater than 15 psig.

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

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

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

Technical Reference(s):	OS1002.02, Rev 56, section 4.41 (page 131)		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8024I 08		
Question Source:	Bank #	x	TEB 34954
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2010 Seabrook NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(7)	

	55.43	
<p>K/A match justification: The design feature that controls hydrogen in the RCS is the overpressure of hydrogen gas in the VCT. This question is testing the students' knowledge of how to properly operate this design feature.</p>		

Examination Outline Cross-reference:	Level	RO	SRO															
Q31	Tier #	2																
	Group #	1																
	K/A #	005 (SF4P RHR) Residual Heat Removal K6.03 Knowledge of the effect of a loss or malfunction on the following will have on the RHRS: RHR heat exchanger.																
	Importance Rating	2.5																
Proposed Question:																		
<p>The following plant conditions exist:</p> <ul style="list-style-type: none"> • The plant is in MODE 5. • Train 'B' RHR is in service in COOLDOWN mode. • Core Exit Thermocouple Temperature is 182 °F and STABLE • RHR HEAT EXCHANGER OUTLET VALVE, RH-HCV-607 is 10% OPEN • RHR HEAT EXCHANGER BYPASS FLOW CONTROL VALVE, RH-FCV-619, is maintaining total RHR flow at 3500 gpm • A loss of Instrument Air pressure occurs. <p>Which of the following describes the effect on the RHR system and on RCS temperature?</p> <table border="0" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="text-align: left; border-bottom: 1px solid black;"><u>RH-HCV-607</u></th> <th style="text-align: left; border-bottom: 1px solid black;"><u>RH-FCV-619</u></th> <th style="text-align: left; border-bottom: 1px solid black;"><u>RCS Temperature</u></th> </tr> </thead> <tbody> <tr> <td>A. FAILS AS IS</td> <td>FAILS AS IS</td> <td>STABLE</td> </tr> <tr> <td>B. FAILS AS IS</td> <td>FAILS CLOSED</td> <td>INCREASES</td> </tr> <tr> <td>C. FAILS OPEN</td> <td>FAILS CLOSED</td> <td>DECREASES</td> </tr> <tr> <td>D. FAILS OPEN</td> <td>FAILS AS IS</td> <td>DECREASES</td> </tr> </tbody> </table>				<u>RH-HCV-607</u>	<u>RH-FCV-619</u>	<u>RCS Temperature</u>	A. FAILS AS IS	FAILS AS IS	STABLE	B. FAILS AS IS	FAILS CLOSED	INCREASES	C. FAILS OPEN	FAILS CLOSED	DECREASES	D. FAILS OPEN	FAILS AS IS	DECREASES
<u>RH-HCV-607</u>	<u>RH-FCV-619</u>	<u>RCS Temperature</u>																
A. FAILS AS IS	FAILS AS IS	STABLE																
B. FAILS AS IS	FAILS CLOSED	INCREASES																
C. FAILS OPEN	FAILS CLOSED	DECREASES																
D. FAILS OPEN	FAILS AS IS	DECREASES																
Proposed Answer:	C.																	
Explanation (Optional):																		

C is correct. A failure of IA to the RHR system will result in RH-HCV-607 failing open and RH-FCV-619 failing closed. This will force full flow through the RHR heat exchanger and cause RCS temperature to decrease.

A is incorrect but plausible. The failure directions (open/closed) of the RHR system valves is a common misconception. A failure of IA to the RHR system will result in RH-HCV-607 failing open and RH-FCV-619 failing closed. This will force full flow through the RHR heat exchanger and cause RCS temperature to decrease. It is plausible that if the valves were to fail as is, temperature would remain stable.

B is incorrect but plausible. The failure directions (open/closed) of the RHR system valves is a common misconception. A failure of IA to the RHR system will result in RH-HCV-607 failing open and RH-FCV-619 failing closed. This will force full flow through the RHR heat exchanger and cause RCS temperature to decrease.

D is incorrect but plausible. The failure directions (open/closed) of the RHR system valves is a common misconception. A failure of IA to the RHR system will result in RH-HCV-607 failing open and RH-FCV-619 failing closed. This will force full flow through the RHR heat exchanger and cause RCS temperature to decrease.

Technical Reference(s):	N/A		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8033I 07		
Question Source:	Bank #	x	TEB 29863
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(7)	

2020 Seabrook Station NRC Written Exam
ES-401-5 Written Examination Question Worksheet

	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO										
Q32	Tier #	2											
	Group #	1											
	K/A #	006 (SF2; SF3 ECCS) Emergency Core Cooling K3.01 Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: RCS											
	Importance Rating	4.1											
Proposed Question:													
<p>The crew is implementing FR-C.2, "Response to Degraded Core Cooling". While isolating the SI accumulators, 'B' SI Accumulator Isolation Valve SI-V-17 will not close.</p> <p>What actions must the crew take in response (1) and why (2)?</p> <table border="0" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="width: 50%; text-align: center;">(1)</th> <th style="width: 50%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A. Vent the 'B' Accumulator.</td> <td>Pressurized nitrogen injection will impede subsequent depressurization.</td> </tr> <tr> <td>B. Vent the 'B' Accumulator.</td> <td>Injected nitrogen will collect in high places causing gas binding and reduced heat transfer in the SG U-tubes.</td> </tr> <tr> <td>C. Do not cooldown to less than 410 °F.</td> <td>Pressurized nitrogen injection will impede subsequent depressurization.</td> </tr> <tr> <td>D. Do not cooldown to less than 410 °F.</td> <td>Injected nitrogen will collect in high places causing gas binding and reduced heat transfer in the SG U-tubes.</td> </tr> </tbody> </table>				(1)	(2)	A. Vent the 'B' Accumulator.	Pressurized nitrogen injection will impede subsequent depressurization.	B. Vent the 'B' Accumulator.	Injected nitrogen will collect in high places causing gas binding and reduced heat transfer in the SG U-tubes.	C. Do not cooldown to less than 410 °F.	Pressurized nitrogen injection will impede subsequent depressurization.	D. Do not cooldown to less than 410 °F.	Injected nitrogen will collect in high places causing gas binding and reduced heat transfer in the SG U-tubes.
(1)	(2)												
A. Vent the 'B' Accumulator.	Pressurized nitrogen injection will impede subsequent depressurization.												
B. Vent the 'B' Accumulator.	Injected nitrogen will collect in high places causing gas binding and reduced heat transfer in the SG U-tubes.												
C. Do not cooldown to less than 410 °F.	Pressurized nitrogen injection will impede subsequent depressurization.												
D. Do not cooldown to less than 410 °F.	Injected nitrogen will collect in high places causing gas binding and reduced heat transfer in the SG U-tubes.												

Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. Step 12 of FR-C.2, checks if the accumulators should be isolated. If RCS hot leg temperatures are less than 410 °F, the accumulators would have injected water contents but not nitrogen gas. The operator is then directed to isolate the accumulators to prevent nitrogen injection in the subsequent depressurization. If the accumulators cannot be isolated, they will be vented per step 12d RNO. Nitrogen injection is prevented as it could collect in the high places and cause gas binding and reduced heat transfer in the SG U-tubes.</p> <p>A is incorrect but plausible. Part (1) is correct. Nitrogen injection is prevented as it could collect in the high places and cause gas binding and reduced heat transfer in the SG U-tubes. It is plausible that the injection of pressurized nitrogen would impeded further depressurization but this is not the basis.</p> <p>C is incorrect but plausible. Accumulators are isolated only when two RCS hot leg temperatures are less than 410 °F per step 12a. It is conceivable that maintaining temperatures above this point would be the required action to prevent nitrogen injection as this temperature is indicative of accumulator injection. Nitrogen injection is prevented as it could collect in the high places and cause gas binding and reduced heat transfer in the SG U-tubes. it is plausible that the injection of pressurized nitrogen would impeded further depressurization but this is not the basis.</p> <p>D is incorrect but plausible. Accumulators are isolated only when two RCS hot leg temperatures are less than 410 °F per step 12a. It is conceivable that maintaining temperatures above this point would be the required action to prevent nitrogen injection as this temperature is indicative of accumulator injection. Part (2) is correct.</p>			
Technical Reference(s):	FR-C.2, "Response to Degraded Core Cooling" rev 27 Background document for FR-C.2, Rev 3		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1227I 09		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive	Memory or Fundamental Knowledge	x	

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 ES-401-5 Written Examination Question Worksheet

Level:	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41			
	55.43	(7)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q33	Tier #	2	
	Group #	1	
	K/A #	007 (SF5 PRTS) Pressurizer Relief/Quench Tank A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the P S; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Abnormal pressure in the PRT	
	Importance Rating	2.6	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A loss of offsite power has occurred • Due to a small break LOCA the crew is performing ES-1.2, "Post LOCA Cooldown and Depressurization". • Normal charging has been established. • Letdown is not in service. • The crew is preparing to depressurize the RCS to minimize subcooling using a PORV. • The PRT rupture disk is intact. • As the crew opens the PORV it is found to be ineffective at decreasing RCS pressure. <p>What is the reason that the PORV is ineffective and what actions should be taken?</p> <p>A. With the PRT rupture disk intact, the PORV loses effectiveness as PZR pressure approaches PRT pressure. Use Aux Spray to perform the depressurization.</p> <p>B. With the PRT rupture disk intact, the PORV loses effectiveness as PZR pressure approaches PRT pressure. Use normal spray to perform the depressurization.</p>			

C. The PORV loses effectiveness when the RCPs are shut down because of the reduced vessel d/p. Use Aux Spray to perform the depressurization.	
D. The PORV loses effectiveness when the RCPs are shut down because of the reduced vessel d/p. Use normal spray to perform the depressurization.	
Proposed Answer:	A.
Explanation (Optional):	
<p>A is correct. From the background document for ES-1.2 “a PRZR PORV may not be effective for RCS depressurization at low temperature and pressure conditions that could exist in the pressurizer when the PRT rupture disk is still intact, and RCS pressure approaches PRT pressure.” This is the reason that the PORV is ineffective under the given conditions. With the PORV ineffective, ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization.</p> <p>B is incorrect but plausible. From the background document for ES-1.2 “a PRZR PORV may not be effective for RCS depressurization at low temperature and pressure conditions that could exist in the pressurizer when the PRT rupture disk is still intact, and RCS pressure approaches PRT pressure.” This is the reason that the PORV is ineffective under the given conditions. With the PORV ineffective, ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization. Normal spray is not available without RCPs running but is a plausible distractor as it is typically the preferred method of depressurization.</p> <p>C is incorrect but plausible. The PORV is has lost effectiveness here because the PRT rupture disk is intact, not because the RCPs are shut down. This is plausible because a reduced d/p renders the pressurizer spray valves ineffective. ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization.</p> <p>D is incorrect but plausible. The PORV is has lost effectiveness here because the PRT rupture disk is intact, not because the RCPs are shut down. This is plausible because a reduced d/p renders the pressurizer spray valves ineffective. ES-1.2 will direct the crew to use auxiliary spray to perform the depressurization.</p>	
Technical Reference(s):	<p>ES-1.2, “POST LOCA Cooldown and Depressurization”.</p> <p>Background document for ES-1.2</p>
Proposed references to be provided to applicants during examination:	None
Learning Objective:	SBK LOP L1204I 03

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 ES-401-5 Written Examination Question Worksheet

Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	x		
Question History:				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41	(5)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO										
Q34	Tier #	2											
	Group #	1											
	K/A #	007 (SF5 PRTS) Pressurizer Relief/Quench Tank K5.02 Knowledge of the operational implications of the following concepts as they apply to PRTS: Method of forming a steam bubble in the PZR											
	Importance Rating	3.1											
Proposed Question:													
<p>When performing a Reactor Coolant System fill and vent per OS1001.01, "Reactor Coolant System Fill and Vent", the reactor vessel head is vented to the <u> (1) </u> and when forming a steam bubble in the pressurizer, the Technical Specification limit on <u>RCS</u> heatup is <u> (2) </u> °F/hr.</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. PRT</td> <td>100</td> </tr> <tr> <td>B. RCDT</td> <td>100</td> </tr> <tr> <td>C. PRT</td> <td>200</td> </tr> <tr> <td>D. RCDT</td> <td>200</td> </tr> </table>				(1)	(2)	A. PRT	100	B. RCDT	100	C. PRT	200	D. RCDT	200
(1)	(2)												
A. PRT	100												
B. RCDT	100												
C. PRT	200												
D. RCDT	200												
Proposed Answer:	A.												
Explanation (Optional):													
<p>A is correct. Per OS1001.01, "Reactor Coolant System Fill and Vent" the reactor vessel head is vented to the PRT to bring RVLIS level to >103%. Technical Specifications 3.4.9.1 for RCS heatup/cool-down limits and 3.4.9.2 for pressurizer heatup/cool-down limits both limit the heatup rate to 100°F/hr.</p> <p>B is incorrect but plausible. Numerous RCS systems drain to the RCDT, e.g. RCP #2 seal leak off, system reliefs and excess letdown. The reactor vessel head however, is vented to the PRT. Part 2 is correct.</p> <p>C is incorrect but plausible. Part 1 is correct. Tech Spec 3.4.9.2 allows for a maximum cool-down rate in the pressurizer of 200 °F/hr, making this a plausible distractor.</p>													

<p>D is incorrect by plausible. Numerous RCS systems drain to the RCDT, e.g. RCP #2 seal leak off, system reliefs and excess letdown. The reactor vessel head however, is vented to the PRT. Tech Spec 3.4.9.2 allows for a maximum cooldown rate in the pressurizer of 200 °F/hr, making this a plausible distractor.</p>			
<p>Technical Reference(s):</p>		<p>Technical Specifications 3.4.9.1 and 3.4.9.2 OS1001.01, "Reactor Coolant System Fill and Vent".</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L8021I 04, 08</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>x</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(5)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q35	Tier #	2	
	Group #	1	
	K/A #	008 (SF8 CCW) Component Cooling Water K4.02 Knowledge of CCWS design feature(s) and/or interlock(s) which provide for the following: Operation of the surge tank, including the associated valves and controls.	
	Importance Rating	2.9	
Proposed Question:			
<p>'A' Train PCCW head tank level is decreasing due to a leak. Which of the following includes automatic actions that will occur when level reaches less than 36%?</p> <p>A. WPB Train 'A' supply valves isolate (CC-V-426 and CC-V-427) and Isolates CC to the Spent Fuel Hx's (CC-V-32)</p> <p>B. WPB Train 'A' supply valves isolate (CC-V-426 and CC-V-427) and Train 'A' Radiation Monitor isolates (CC-V-975 and CC-V-1298)</p> <p>C. Train 'A' Thermal Barrier Supply Valves isolate (CC-V-1101 and CC-V-1109)</p> <p>D. Loop 'A' PCCW supply valves to containment isolate (CC-V-168, 57,121,122)</p>			
Proposed Answer:	D.		
Explanation (Optional):			
<p>D is correct. PCCW to containment will get isolated at <36% PCCW head tank level.</p> <p>A is incorrect but plausible. At 42% level in the PCCW head tank, the PCCW supply to the WPB and the Rad monitor will isolate. The SF heat exchangers isolate on a 'T' signal. This is a common</p>			

misconception.			
B is incorrect but plausible. At 42% level in the PCCW head tank, the PCCW supply to the WPB and the Rad monitor will isolate. The isolation is train related.			
C is incorrect but plausible. The thermal barrier system does not get isolated on low PCCW head tank level. This is a common misconception.			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8036I 12		
Question Source:	Bank #	X	TEB 32174
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q36	Tier #	2	
	Group #	1	
	K/A #	010 (SF3 PZR PCS) Pressurizer Pressure Control 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:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A Turbine Setback has occurred. • The crew is stabilizing the plant. • Indications are that the PORVs had opened. • Pressurizer pressure spiked to 2390 psig and is now 2300 psig and lowering. • The reactor and turbine are online. <p>What action is required?</p> <p>A. Trip the reactor, enter E-0, "Reactor Trip or Safety Injection"</p> <p>B. Stabilize the plant. Adjust control rod position to control AFD.</p> <p>C. Trip the reactor, enter FR-S.1, "Response to Nuclear Power Generation/ATWS"</p> <p>D. Verify the PORVs have closed. Monitor RCS pressure for reactor trip and safety injection set points.</p>			
Proposed Answer:	A		
Explanation (Optional):			

<p>A is correct. Reactor trip set point for pressurizer pressure of 2385 psig was exceeded. The reactor should have tripped and failed to trip automatically. The crew is required to trip the reactor. Once the reactor is tripped, enter E-0.</p>			
<p>B is incorrect but plausible. This action would normally be required following a large down power.</p>			
<p>C is incorrect but plausible. Failure of the reactor to trip automatically is not an entry condition for FR-S.1 if the reactor can be manually tripped. FR-S.1 is only entered from E-0 after a manual reactor trip is attempted.</p>			
<p>D is incorrect but plausible. With RCS pressure decreasing but above 2235 psig, continued monitoring for reactor trip and safety injection is plausible.</p>			
<p>Technical Reference(s):</p>		<p>FR-S.1, "Response to Nuclear Power Generation/ATWS" Rev 30</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1200I 15</p>		
<p>Question Source:</p>	<p>Bank #</p>	<p>X</p>	<p>11946</p>
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>		
<p>Question History:</p>	<p>2008 Indian Point NRC Exam</p>		
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		
	<p>Comprehension or Analysis</p>		<p>X</p>
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q37	Tier #	2	
	Group #	1	
	K/A #	010 (SF3 PZR PCS) Pressurizer Pressure Control K6.01 Knowledge of the effect of a loss or malfunction of the following will have on the PZR PCS: Pressure detection systems	
	Importance Rating	2.7	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power, Normal operating pressure and temperature. • Pressurizer pressure channels are selected to 457/456 for control and backup. • Pressurizer pressure channel PT-457 fails LOW. <p>With <u>no operator action</u> how will the plant respond?</p> <p>A. <u>All</u> pressurizer heaters will energize and <u>no</u> PORV will open.</p> <p>B. <u>Only</u> the control group heaters will energize and <u>no</u> PORV will open.</p> <p>C. <u>All</u> pressurizer heaters will energize and <u>only</u> the 'A' PORV will open.</p> <p>D. <u>Only</u> the control group heaters will energize and <u>only</u> the 'A' PORV will open.</p>			
Proposed Answer:	A.		
Explanation (Optional):			
<p>A is correct. All pressurizer heaters will receive a demand signal to energize. No PORV will open with 457 selected for control after it fails low. In this alignment channel 457 arms the 'B' PORV and causes the demand signal to open the 'A' PORV.</p> <p>B is incorrect but plausible. All pressurizer heaters will receive a demand signal to energize. it is plausible that only the control group heaters receive the demand to energize from the controlling</p>			

<p>channel failing low. No PORV will open with 457 selected for control after it fails low. In this alignment channel 457 arms the 'B' PORV and causes the demand signal to open the 'A' PORV.</p> <p>C is incorrect but plausible. All pressurizer heaters will receive a demand signal to energize. No PORV will open with 457 selected for control after it fails low. In this alignment channel 457 arms the 'B' PORV and causes the demand signal to open the 'A' PORV. It is plausible that the 'A' PORV opens as it is normally controlled from channel 455.</p> <p>D is incorrect but plausible. All pressurizer heaters will receive a demand signal to energize. No PORV will open with 457 selected for control after it fails low. In this alignment channel 457 arms the 'B' PORV and causes the demand signal to open the 'A' PORV. It is plausible that the 'A' PORV opens as it is normally controlled from channel 455.</p>			
Technical Reference(s):		OS1201.06, "PZR Pressure Instrument/Component Failure" Rev 15	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8027I 14		
Question Source:	Bank #	<input type="checkbox"/>	<input type="checkbox"/>
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input checked="" type="checkbox"/>	<input type="checkbox"/>
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		<input type="checkbox"/>
	Comprehension or Analysis		<input checked="" type="checkbox"/>
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO										
Q38	Tier #	2											
	Group #	1											
	K/A #	012 (SF7 RPS) Reactor Protection K5.01 Knowledge of the operational implications of the following concepts as they apply to the RPS: DNB											
	Importance Rating	3.3											
Proposed Question:													
<p>The ____ (1) ____ reactor trip provides core protection from departure from nucleate boiling. The trip setpoint is automatically reduced when RCS pressure ____ (2) ____.</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. Overpower ΔT</td> <td>rises</td> </tr> <tr> <td>B. Overtemperature ΔT</td> <td>rises</td> </tr> <tr> <td>C. Overpower ΔT</td> <td>lowers</td> </tr> <tr> <td>D. Overtemperature ΔT</td> <td>lowers</td> </tr> </table>				(1)	(2)	A. Overpower ΔT	rises	B. Overtemperature ΔT	rises	C. Overpower ΔT	lowers	D. Overtemperature ΔT	lowers
(1)	(2)												
A. Overpower ΔT	rises												
B. Overtemperature ΔT	rises												
C. Overpower ΔT	lowers												
D. Overtemperature ΔT	lowers												
Proposed Answer:	D.												
Explanation (Optional):													
<p>D is correct. Technical Specifications bases page B 2-5 gives the basis for reactor trips. The basis for the OTDT trip is to prevent DNB. As RCS pressure decreases the trip setpoint is lowered to prevent DNB.</p> <p>A is incorrect but plausible. The basis of the OPDT trip is to provide assurance of fuel integrity not prevent exceeding DNB.</p> <p>B is incorrect but plausible. The basis for the OTDT trip is to prevent DNB, however as RCS pressure rises the trip setpoint will be increased.</p> <p>C is incorrect but plausible. The basis of the OPDT trip is to provide assurance of fuel integrity not</p>													

prevent exceeding DNB.			
Technical Reference(s):		Technical Specifications page B 2-5	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8056I 18		
Question Source:	Bank #	x	12912
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2011 Turkey Point NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q39		Tier #	2	
		Group #	1	
		K/A #	013 (SF2 ESFAS) Engineered Safety Features Actuation 2.2.22 Knowledge of limiting conditions for operations and safety limits.	
		Importance Rating	4.0	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> All control rods are inserted. RCS temperature is 400 °F and lowering. Due to multiple control and safety system failures RCS pressure is 2785 psig. <p>Which of the following actions is required in accordance with Tech Specs section 2.0, Safety Limits and Limiting Safety System Settings?</p> <p>A. Restore RCS pressure to ≤ 2735 psig within 5 minutes and be in mode 4 within 1 hour. B. Restore RCS pressure to ≤ 2185 psig within 5 minutes and be in mode 4 within 1 hour. C. Restore RCS pressure to ≤ 2735 psig within 5 minutes only. D. Restore RCS pressure to ≤ 2185 psig within 5 minutes only.</p>				
Proposed Answer:	C.			
Explanation (Optional):				
<p>C is correct. Safety limit 2.1.2 requires RCS pressure be maintained ≤ 2735 psig. TS 2.1.3 requires that if safety limit 2.1.2 is violated in mode 3, compliance be restored within 5 minutes.</p> <p>A is incorrect but plausible. 2735 psig is the correct safety limit pressure, however there is no requirement to be in mode 4 within 1 hour. There is a requirement if the plant were in mode 1 or 2</p>				

to be in mode 3 within 1 hour, making this part of the distracter plausible.				
B is incorrect but plausible. 2185 psig is not the safety limit on RCS pressure, it is the DNB lower limit on RCS pressure. Safety limit 2.1.2 requires RCS pressure be maintained \leq 2735 psig. TS 2.1.3 requires that if safety limit 2.1.2 is violated in mode 3, compliance be restored within 5 minutes.				
D is incorrect but plausible. 2185 psig is not the safety limit on RCS pressure, it is the DNB lower limit on RCS pressure.				
Technical Reference(s):		Technical Specifications page 2-1.		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8010I 04			
Question Source:	Bank #			
	Modified Bank#	x	12030	(Note changes or attach Parent)
	New			
Question History:	2009 Seabrook NRC Exam			
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41	(5)		

	55.43	
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Comments: Original question:

The following sequence of events occurs:

- The plant is at 100% power
- A 50% load rejection occurs
- Multiple control and safety systems have failed
- RCS pressure has increased to 2785 psig.

Which of the following correctly completes the tech spec statement listing all required actions in accordance with tech spec section 2.0, safety limits and limiting safety system settings?

Restore RCS pressure to less than _____.

- A. 2385 psig within 5 minutes.
- B. 2735 psig within 5 minutes.
- C. 2385 psig immediately and be in HOT STANDBY within 1 hour.
- D. 2735 psig immediately and be in HOT STANDBY within 1 hour.

Correct answer D.

Examination Outline Cross-reference:	Level	RO	SRO
Q40	Tier #	2	
	Group #	1	
	K/A #	022 (SF5 CCS) Containment Cooling K1.01 Knowledge of the physical connections and/or cause effect relationships between the CCS and the following systems: SWS/cooling system	
	Importance Rating	3.5	
Proposed Question:			
<p>Which condition will result in an automatic trip of the Containment Structure Cooling fans?</p> <p>A. Safety Injection (S) Signal.</p> <p>B. Low PCCW flow to cooling coil <150gpm.</p> <p>C. Containment pressure at the Hi-1 setpoint.</p> <p>D. Containment temperature greater than 135°F.</p>			
Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. Low PCCW flow of less than 150 gpm will cause a trip of the containment structure cooling fans, 1-CAH-FN-1A-F. The flow is sensed by a swatch at the fan.</p> <p>A is incorrect but plausible. A combination of an SI with a LOP will result in the CAH fans being block from automatically restarting, however the SI will not trip the fans by itself. This is a common misconception.</p> <p>C is incorrect but plausible. Containment pressure at Hi-1 (4.3 psig) will cause an SI signal to be generated. The SI will not result in the fans tripping. PCCW to containment will be isolated on a 'P' signal which will trip the CAH fans on low PCCW flow. This does not occur on a SI signal though.</p> <p>D is incorrect but plausible. 135 °F is the high temperature trip setpoint for the PCCW pumps. The pumps will trip if PCCW return temperature is >135 °F on 2/2 instruments for >60 seconds.</p>			

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Technical Reference(s):	N/A		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8038I 04		
Question Source:	Bank #	X	TEB 6551
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(2)-(9)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q41		Tier #	2	
		Group #	1	
		K/A #	026 (SF5 CSS) Containment Spray K2.01 Knowledge of bus power supplies to the following: Containment spray pumps	
		Importance Rating	3.4	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • DBA LOCA coincident with a loss of off-site power (SI/LOP). • The following alarms occur: <ul style="list-style-type: none"> • D2329, "Bus E5 UAT INC LN BKR TRIP & L/O" <p>Which of the following will power the CBS pumps, if anything?</p> <p><u>'A' CBS</u> <u>'B' CBS</u></p> <p>A. 'A' EDG SEPS</p> <p>B. 'A' EDG 'B' EDG</p> <p>C. None SEPS</p> <p>D. None 'B' EDG</p>				
Proposed Answer:	D.			
Explanation (Optional):				
<p>D is correct. With the alarms as given, Bus 5 is unavailable The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the 'A' EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the 'A' CBS pump has no available power source. The DG 'B' will power Bus 6 and the 'B' CBS pump. SEPS would be used to power Bus 6 if the 'B' EDG were unavailable; however SEPS is not used to power the CBS pump because of their large load rating.</p>				

<p>A is incorrect but plausible. The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the 'A' EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the 'A' CBS pump has no available power source. The DG 'B' will power Bus 6 and the 'B' CBS pump.</p>			
<p>B is incorrect but plausible. The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the 'A' EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the 'A' CBS pump has no available power source. The DG 'B' will power Bus 6 and the 'B' CBS pump.</p>			
<p>C is incorrect but plausible. The UAT trip and lock out is commonly mistaken to be that only the UAT is unavailable and that the 'A' EDG would be a potential power source for components on Bus 5. However, the entire bus is locked out and no power sources can be aligned. Hence the 'A' CBS pump has no available power source. The DG 'B' will power Bus 6 and the 'B' CBS pump.</p>			
Technical Reference(s):		D6329 and D6608 VPROs	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8020I 08		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q42	Tier #	2	
	Group #	1	
	K/A #	061 (SF4S AFW) Auxiliary/Emergency Feedwater System K2.02 Knowledge of bus power supplies to the following: AFW electric drive pumps	
	Importance Rating	3.7	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 30% power. • MS-V-393, "SG 'A' Main Steam to EFW Pump" is danger tagged closed for repairs. • A reactor trip occurs coincident with a loss of off-site power. • All SG narrow range levels shrink to 15%. • The following alarms occur: <ul style="list-style-type: none"> • D6608, "DG B Lube Oil Pressure Low" • No operator actions have occurred. <p>What EFW pumps if any, will be running?</p> <p>A. None.</p> <p>B. FW-P-37B, "Motor Driven EFW Pump" <u>only</u>.</p> <p>C. FW-P-37A, "Steam Driven EFW Pump" <u>only</u>.</p> <p>D. FW-P-37A, "Steam Driven EFW Pump" <u>and</u> FW-P-37B, "Motor Driven EFW Pump".</p>			
Proposed Answer:	C.		
Explanation (Optional):			
C is correct. With the 'B' EDG low LO pressure alarm the engine will trip and cannot be started. With the B EDG unavailable and a loss of offsite power, Bus 6 will be deenergized without operator action and the Motor Driven EFW pump will not be running. Later actions will align SEPS			

to Bus 6. The steam driven EFW pump will start when both MS-V-394 opens from an 'A' and 'B' SSPS signal. MS-V-394 is a dual train valve. The loss of Bus 6 will not prevent MS-V-394 from opening, this is a common misconception.

A is incorrect but plausible. With the 'B' EDG low LO pressure alarm the engine will trip and cannot be started. With the B EDG unavailable and a loss of offsite power, Bus 6 will be deenergized without operator action and the Motor Driven EFW pump will not be running. It is plausible that a loss of Bus 6 would result in MS-V-394 not opening. This is not the case as SSPS has redundant power supplies from uninterruptable 120 VAC. 15% is less than the 20% EFW actuation setpoint.

B is incorrect but plausible. If the student does not understand the requirement to place SEPS in service with manual operator action, it may be assumed that SEPS is supplying Bus 6 and the Motor Driven EFW pump is running. The student must demonstrate knowledge of the power supplies to the Motor Driven EFW pump.

D is incorrect but plausible. The steam driven EFW pump will start when both MS-V-394 opens from an 'A' and 'B' SSPS signal. MS-V-394 is a dual train valve. The loss of Bus 6 will not prevent MS-V-394 from opening, this is a common misconception. If the student does not understand the requirement to place SEPS in service with manual operator action, it may be assumed that SEPS is supplying Bus 6 and the Motor Driven EFW pump is running. The student must demonstrate knowledge of the power supplies to the Motor Driven EFW pump.

Technical Reference(s):	N/A		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8045I 03		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(7)	

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	55.43	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q43	Tier #	2	
	Group #	1	
	K/A #	039 (SF4S MSS) Main and Reheat Steam A4.04 Ability to manually operate and/or monitor in the control room: Emergency feedwater pump turbines	
	Importance Rating	3.8	
Proposed Question:			
<p>How do the Turbine Driven EFW Pump steam supply valves respond to an EFW actuation signal?</p> <div style="border: 1px solid black; padding: 10px; margin: 10px auto; width: fit-content;"> <ul style="list-style-type: none"> ➤ MS-V393, SG 'A' Main Steam to Emergency Feedwater Pump. ➤ MS-V394, SG 'B' Main Steam to Emergency Feedwater Pump. ➤ MS-V395, Main Steam to Emergency Feedwater Pump. </div> <p>A. MS-V393, MS-V394, <u>and</u> MS-V395 will open 28 seconds after receipt of the EFW actuation signal.</p> <p>B. MS-V393 and MS-V394 will immediately open. MS-V395 will open 28 seconds after receipt of the EFW actuation signal.</p> <p>C. MS-V393 and MS-V394 will immediately open. MS-V395 will open 28 seconds after either MS-V393 or MS-V394 is fully open.</p> <p>D. MS-V393 and MS-V394 will open within 28 seconds of actuation. MS-V395 will open as soon as either MS-V393 or MS-V394 is fully open.</p>			
Proposed Answer:	C.		
Explanation (Optional):			
C is correct. Open limit switch on MS-V-393 or MS-V-394 will cause MS-V-395 to auto open after a 28 second time delay when 395 switch left in closed. Drains are up stream of MS-V-395. This is to ensure adequate moisture removal.			

A is incorrect but plausible. Only MS-V-395 control circuit has the 28 second time delay.			
B is incorrect but plausible. MS-V-395 will auto open after a 28 second time delay when either 393 or 394 are full open. Limit switches start 28 second timer not the actuation signal.			
D is incorrect but plausible. MS-V-395 will auto open after a 28 second time delay when either 393 or 394 are full open. The EFW actuation signal causes 393 & 394 to auto open.			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8045I 04 RO		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 6580
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	<input checked="" type="checkbox"/>	
	Comprehension or Analysis	<input type="checkbox"/>	
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q44		Tier #	2	
		Group #	1	
		K/A #	039 (SF4S MSS) Main and Reheat Steam K3.05 Knowledge of the effect that a loss or malfunction of the MRSS will have on the following: RCS	
		Importance Rating	3.6	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • An electrical grid disturbance causes a trip of the main turbine. • All 345 kV line voltages decrease to zero volts. • All other plant systems and components respond as designed. • Assume no operator action. <p>Where will reactor coolant temperature stabilize and why?</p> <p>A. 557 °F due to Condenser Steam Dump operation. B. 557 °F due to Atmospheric Steam Dump operation. C. 561 °F due to Atmospheric Steam Dump operation. D. 567 °F due to Main Steam Safety Valve operation.</p>				
Proposed Answer:		C.		
Explanation (Optional):				
<p>C is correct. The Main Steam Dumps are not available due to loss of offsite power. 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. MSSVs should not lift. Lowest setpoint of SG safeties = 1185 psig which corresponds to 567°F.</p>				

<p>A is incorrect but plausible. 557°F is the normal plant no load temperature controlled by the steam dumps. However, with the LOP the condenser is not available due to no CW pumps running.</p> <p>B is incorrect but plausible. 557°F is the normal plant no load temperature controlled by the steam dumps. However, with the LOP the condenser is not available due to no CW pumps running. The ASDVs will control temperature at 561°F.</p> <p>D is incorrect but plausible. MSSVs should not lift. Lowest set safeties = 1185 psig which corresponds to 567°F.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8041I 03		
Question Source:	Bank #	x	TEB 30005
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q45		Tier #	2	
		Group #	1	
		K/A #	059 (SF4S MFW) Main Feedwater A1.07 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MFW controls including: Feed Pump speed, including normal control speed for ICS	
		Importance Rating	2.5	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 17% power. • Steam Dump MODE Selector is in the STEAM PRESSURE MODE. • Main Feed Pump 32A is operating in AUTO. • MS-PK-507 is in AUTOMATIC. • Main Steam Header Pressure Instrument PT-507 fails HIGH. <p>How will the 'A' Main Feed Pump Speed and Steam Dumps respond?</p> <p>A. 'A' Main Feed Pump speed will DECREASE. Steam Dumps will OPEN. B. 'A' Main Feed Pump speed will INCREASE. Steam dumps will OPEN. C. 'A' Main Feed Pump speed will DECREASE. Steam dumps will CLOSE. D. 'A' Main Feed Pump speed will INCREASE. Steam dumps will CLOSE.</p>				
Proposed Answer:	B.			
Explanation (Optional):				
B is correct. MS-PT-507 measures main steam header pressure. When PT-507 fails high, the				

<p>main feed water pumps increase speed and the steam dumps, because they are in steam pressure mode in automatic, will open. The feed water pump speed control is based upon maintaining a dP between the common feed water header (FW-PT-508) and the main steam common header (MS-PT-507). This dependence upon PT-507 and 508 is a source of common misconception.</p> <p>A is incorrect but plausible. The relationship between PT-507 508, main feed water pump speed and steam dumps is a common misconception.</p> <p>C is incorrect but plausible. The relationship between PT-507 508, main feed water pump speed and steam dumps is a common misconception.</p> <p>D is incorrect but plausible. The relationship between PT-507 508, main feed water pump speed and steam dumps is a common misconception.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1193I 07		
Question Source:	Bank #	x	23148
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q46	Tier #	2	
	Group #	1	
	K/A #	061 (SF4S AFW) Auxiliary/Emergency Feedwater 2.4.31 Knowledge of annunciator alarms, indications, or response procedures.	
	Importance Rating	4.2	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Reactor trip from 100% power. • ES-0.1, "Reactor Trip Response" is being implemented. • EFW to 'A' and 'D' SGs has been throttled. • EFW to 'B' and 'C' SGs has automatically isolated. • "EFW Flow HIGH" VAS alarms are actuated for all four SGs. • Feed and steam line integrity has been verified per the VPROs. • EFW flow to 'A' and 'D' SG has been verified < 510 gpm each. <p>What actions are necessary to restore EFW to 'B' and 'C' SGs?</p> <p>A. Momentarily place the non-isolated Train 'A' EFW valve switches to THROTTLE OPEN. Restore flow to 'B' and 'C' SGs as required.</p> <p>B. Momentarily place the non-isolated Train 'B' EFW valve switches to THROTTLE CLOSE. Restore flow to 'B' and 'C' SGs as required.</p> <p>C. Momentarily place the non-isolated Train 'A' <u>and</u> 'B' EFW valve switches to THROTTLE OPEN. Restore flow to 'B' and 'C' SGs as required.</p> <p>D. Momentarily place the non-isolated Train 'A' <u>and</u> 'B' EFW valve switches to THROTTLE CLOSE. Restore flow to 'B' and 'C' SGs as required.</p>			
Proposed Answer:	C.		

Explanation (Optional):			
<p>C is correct. This answer correctly summarizes the VPROs for the “SG A (B, C, D) EFW FLOW HIGH” alarms (F5280, F5281, F5449 and F5453). EFW isolation to two SGs as described in the stem means both a Train ‘A’ and ‘B’ isolation signal occurred. Thus, both trains must be reset. Reset is accomplished by momentarily placing the non-isolated Train ‘A’ <u>and</u> ‘B’ EFW valve switches to THROTTLE OPEN, <u>not</u> THROTTLE CLOSE. The isolated valves will reset as soon as the operator restores flow.</p> <p>A is incorrect but plausible. Must reset both trains.</p> <p>B is incorrect but plausible. Must reset both trains. Wrong switch position specified.</p> <p>D is incorrect but plausible. Wrong switch position specified.</p>			
Technical Reference(s):		VPOR for F5280	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8045I 06		
Question Source:	Bank #	x	TEB 31605
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(10)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO										
Q47		Tier #	2											
		Group #	1											
		K/A #	062 (SF6 ED AC) AC Electrical Distribution A1.01 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Significance of D/G load limits											
		Importance Rating	3.4											
Proposed Question:														
<p>The crew is performing OX1426.01, "DG 1A Monthly Operability Surveillance"</p> <p>The DG 1A output breaker has been closed and the crew is preparing to load the engine.</p> <p>The operator takes the DG A SPEED ADJUST switch to raise.</p> <p>What parameter will change (1) and what limit must not be exceeded while loading the engine (2)?</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. MW</td> <td>6083 MW continuous rating</td> </tr> <tr> <td>B. MW</td> <td>6083 MW for 168 hours/year.</td> </tr> <tr> <td>C. kVAR</td> <td>4014 kVAR.</td> </tr> <tr> <td>D. kVAR</td> <td>4866 kVAR.</td> </tr> </table>					(1)	(2)	A. MW	6083 MW continuous rating	B. MW	6083 MW for 168 hours/year.	C. kVAR	4014 kVAR.	D. kVAR	4866 kVAR.
(1)	(2)													
A. MW	6083 MW continuous rating													
B. MW	6083 MW for 168 hours/year.													
C. kVAR	4014 kVAR.													
D. kVAR	4866 kVAR.													
Proposed Answer:		A.												
Explanation (Optional):														
A is correct. Taking the speed adjust switch to raise with the output breaker closed during this surveillance (operating in parallel with offsite power) will cause MW loading to increase. The continuous operating limit is 6083 MW.														

B is incorrect but plausible. Taking the speed adjust switch to raise with the output breaker closed during this surveillance (operating in parallel with offsite power) will cause MW loading to increase. The continuous operating limit is 6083 MW the 168 hour/year is 6697 MW.

C is incorrect but plausible. Adjusting the voltage output will adjust kVAR not MW. This is a common misconception. 4014 kVAR is approximately $\frac{2}{3} * 6083$ where $\frac{2}{3}$ is the ratio of kVARs to MW loading that is normally maintained.

D is incorrect but plausible. Adjusting the voltage output will adjust kVAR not MW. This is a common misconception. 4866 kVAR is approximately $0.8 * 6083$ where 0.8 is the limiting power factor for DG operation.

Technical Reference(s):	OX1426.01, "DG 1A monthly Operability Surveillance" Rev 49		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8020I 02		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q48	Tier #	2	
	Group #	1	
	K/A #	062 (SF6 ED AC) AC Electrical Distribution A1.03 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the ac distribution system controls including: Effect on instrumentation and controls of switching power supplies	
	Importance Rating	2.5	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • PP-1D fails and becomes de-energized. • OS1247.01, "Loss of a 120 VAC Vital Instrument Panel (PP-1A, 1B, 1C or 1D)" is being implemented. <p>What is the impact on Train 'B' SSPS and what action is required per the AOP?</p> <p>A. The redundant power supply to the logic cards is lost ONLY. Restore power to PP-1D from its maintenance supply ONLY.</p> <p>B. The redundant power supply to the logic cards is lost ONLY. Restore power to PP-1D from its maintenance supply AND reset the power supply.</p> <p>C. The redundant power supply to the logic cards AND the MCB Demultiplexer is lost. Restore power to PP-1D from its maintenance supply ONLY.</p> <p>D. The redundant power supply to the logic cards AND the MCB Demultiplexer is lost. Restore power to PP-1D from its maintenance supply AND reset the power supply.</p>			

Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. PP-1D only provides a redundant power supply to the logic bay of Train 'B' SSPS. Per OS 1247.01, the proper crew response is to restore power to PP-1D using the maintenance supply and reset Train 'B' SSPS power supply.</p> <p>A is incorrect but plausible. Since only the redundant power supply to the logic bay is lost, the student could conclude that resetting the SSPS power supply is not required.</p> <p>C is incorrect but plausible. PP-1D and PP-1B provide power to the logic bay. However, only PP-1C provides power to the MCB Demultiplexer. This is a common student mistake. Since the logic bay remains powered, the student could conclude that resetting the SSPS power supply is not required.</p> <p>D is incorrect but plausible. PP-1D and PP-1B provide power to the logic bay. However, only PP-1C provides power to the MCB Demultiplexer. This is a common student mistake.</p>			
Technical Reference(s):	OS1247.01, "Loss of a 120 VAC Vital Instrument Panel"		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1186I 09		
Question Source:	Bank #	x	TEB 31635
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5)	

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Comments:		

Examination Outline Cross-reference:	Level	RO	SRO															
Q49	Tier #	2																
	Group #	1																
	K/A #	063 (SF6 ED DC) DC Electrical Distribution A4.03 Ability to manually operate and/or monitor in the control room: Battery discharge rate																
	Importance Rating	3.0																
Proposed Question:																		
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • Electric distribution system is in normal alignment. • No maintenance is in progress. • VAS Alarm D6066 "DC Bus 11C Grounded" is received. • The ground is subsequently located and removed from the bus. <p>What action(s) is/are necessary in accordance with the alarm response procedure to reset the ground alarm (1) and where can the <u>charging/discharging</u> status of the vital batteries be monitored (2)?</p> <table style="width: 100%; border: none;"> <thead> <tr> <th style="width: 5%;"></th> <th style="width: 65%; text-align: center;">(1)</th> <th style="width: 30%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A.</td> <td>Depress the Ground Reset push-buttons at Bus 11C <u>and</u> on MCB-HR.</td> <td>Local digital ammeters.</td> </tr> <tr> <td>B.</td> <td>Depress the Ground Reset push-button on MCB-HR <u>only</u>.</td> <td>Local digital ammeters.</td> </tr> <tr> <td>C.</td> <td>Depress the Ground Reset push-buttons at Bus 11C <u>and</u> on MCB-HR.</td> <td>Ammeters on MCB-HR.</td> </tr> <tr> <td>D.</td> <td>Depress the Ground Reset push-button on MCB-HR <u>only</u>.</td> <td>Ammeters on MCB-HR.</td> </tr> </tbody> </table>					(1)	(2)	A.	Depress the Ground Reset push-buttons at Bus 11C <u>and</u> on MCB-HR.	Local digital ammeters.	B.	Depress the Ground Reset push-button on MCB-HR <u>only</u> .	Local digital ammeters.	C.	Depress the Ground Reset push-buttons at Bus 11C <u>and</u> on MCB-HR.	Ammeters on MCB-HR.	D.	Depress the Ground Reset push-button on MCB-HR <u>only</u> .	Ammeters on MCB-HR.
	(1)	(2)																
A.	Depress the Ground Reset push-buttons at Bus 11C <u>and</u> on MCB-HR.	Local digital ammeters.																
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D.	Depress the Ground Reset push-button on MCB-HR <u>only</u> .	Ammeters on MCB-HR.																

Proposed Answer:	B.	
Explanation (Optional):		
<p>B is correct. DC bus ground alarms must be reset from the MCB section HR. Local digital ammeters indicate the battery discharge rate or charging rate.</p> <p>A is incorrect but plausible. It is not required to reset a DC bus ground alarm locally at the DC bus. This is testing the student's ability to operate the DC distribution system in response to ground conditions. Local digital ammeters indicate the battery discharge rate or charging rate.</p> <p>C is incorrect but plausible. Using VAS alarm reset pushbuttons is the normal means of resetting alarms. However, for this ground condition the ground alarms must be reset from the MCB section HR. Ammeters on the MCB only indicate ground current.</p> <p>D is incorrect but plausible. DC bus ground alarms must be reset from the MCB section HR. Ammeters on the MCB only indicate ground current.</p>		
Technical Reference(s):	VPRO for D6066	
Proposed references to be provided to applicants during examination:	None	
Learning Objective:	SBK LOP L8017I 07	
Question Source:	Bank #	
	Modified Bank#	(Note changes or attach Parent)
	New	x
Question History:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	(7)

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Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q50	Tier #	2	
	Group #	1	
	K/A #	064 (SF6 EDG) Emergency Diesel Generator K1.01 Knowledge of the physical connections and/or cause effect relationships between the ED/G system and the following systems: AC distribution system	
	Importance Rating	4.1	
Proposed Question:			
<p>Which event will result in the <u>immediate</u> actuation of the Bus E5 Emergency Power Sequencer?</p> <p>A. 345 kV Bus 5 is de-energized due to a fault on the non-segregated bus duct. Bus E5 voltage just dipped to 2700 volts.</p> <p>B. 345 kV Bus 5 is de-energized due to a fault on the non-segregated bus duct. Bus E5 voltage just dipped to 3900 volts.</p> <p>C. The 4.16 kV distribution system is in its normal Mode 1 configuration with no faults on any bus. A Safety Injection signal was actuated 13 seconds ago. Bus E5 voltage just dipped to 3000 volts.</p> <p>D. The 4.16 kV distribution system is in its normal Mode 1 configuration with no faults on any bus. Bus E5 voltage has been 3900 volts for 15 seconds. A Safety Injection signal was just actuated.</p>			
Proposed Answer:	A.		
Explanation (Optional):			
<p>The EPS is activated only during a sustained loss of power to its emergency bus, as determined by either (1) emergency bus first-level undervoltage protection (less than 70% of nominal voltage for 1.2 seconds), or (2) emergency bus second-level undervoltage protection (less than 95% of nominal voltage with coincident SI signal for 10 seconds). The first-level undervoltage time delay of 1.2 seconds allows time for the bus to be automatically transferred to the alternate (RAT) supply, if possible. However, if off-site power is not available, as sensed on the nonsegregated bus duct between the RAT and the RAT incoming supply breaker, the EPS is activated immediately upon sensing the undervoltage condition on the bus.</p>			

<p>A is correct. First level undervoltage will activate the EPS immediately because loss of 345 kV bus 5 renders the RAT unavailable. 2700 V is less than 70% of nominal ($2700 \text{ V} / 4160 \text{ V} = 0.64$)</p> <p>B is incorrect but plausible. 3900 V is not less than 70% of nominal ($3900 \text{ V} / 4160 \text{ V} = 0.93$), but is less than the 95% requirement for second level undervoltage.</p> <p>C is incorrect but plausible. Both the SI and low voltage condition must be met for 10 seconds in order to activate EPS on the second level undervoltage. This is a common misconception.</p> <p>D is incorrect but plausible. Both the SI and low voltage condition must be met for 10 seconds in order to activate EPS on the second level undervoltage. This is a common misconception.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8020I 08		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 19948
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	<input checked="" type="checkbox"/>	
	Comprehension or Analysis	<input type="checkbox"/>	
10 CFR Part 55 Content:	55.41	(2)-(9)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO										
Q51	Tier #	2											
	Group #	1											
	K/A #	073 (SF7 PRM) Process Radiation Monitoring A2.02 Ability to (a) predict the impacts of the following malfunctions or operations on the PRM system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Detector failure											
	Importance Rating	2.7											
Proposed Question:													
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • RM-6504, "WG Compressor Discharge Rad Monitor" goes into ALARM. <p>What automatic action will occur (1), and if this action fails to occur, what action is required in accordance with OS1252.01, "Process or Effluent High Radiation" (2)?</p> <p>WG-FV-1602: "Waste Gas to F-16". VG-V-57: "PAB Hydrogenated Vent Header Isolation".</p> <table style="width: 100%; border: none;"> <tr> <td style="text-align: center; width: 50%;">(1)</td> <td style="text-align: center; width: 50%;">(2)</td> </tr> <tr> <td>A. WG-FV-1602 will close</td> <td>Close VG-V-57 at CP-38</td> </tr> <tr> <td>B. WG-FV-1602 will close</td> <td>Close WG-FV-1602 at MCB-CR</td> </tr> <tr> <td>C. VG-V-57 will close</td> <td>Close VG-V-57 at CP-38</td> </tr> <tr> <td>D. VG-V-57 will close</td> <td>Close WG-FV-1602 at MCB-CR</td> </tr> </table>				(1)	(2)	A. WG-FV-1602 will close	Close VG-V-57 at CP-38	B. WG-FV-1602 will close	Close WG-FV-1602 at MCB-CR	C. VG-V-57 will close	Close VG-V-57 at CP-38	D. VG-V-57 will close	Close WG-FV-1602 at MCB-CR
(1)	(2)												
A. WG-FV-1602 will close	Close VG-V-57 at CP-38												
B. WG-FV-1602 will close	Close WG-FV-1602 at MCB-CR												
C. VG-V-57 will close	Close VG-V-57 at CP-38												
D. VG-V-57 will close	Close WG-FV-1602 at MCB-CR												

Proposed Answer:	B.	
Explanation (Optional):		
<p>B is correct. With RM-6504 in alarm, WG-FV-1602 should automatically close to isolate waste gas to filter 16. If this fails to occur automatically, abnormal procedure OS1252.01 will direct the crew to manually close the valve via the control switch on MCB-CR.</p> <p>A is incorrect but plausible. VG-V-57 is the WG header isolation to filter 16. It is interlocked with filter 16 fans, PAH-FN-8 A and B. It is not interlocked with the WG radiation monitors. This is a common misconception. The valve is operated at CP-38. If 1602 fails to automatically close OS1252.01 will direct manually closing of it from the MCB, not VG-V-57 from CP-38.</p> <p>C is incorrect but plausible. VG-V-57 is the WG header isolation to filter 16. It is interlocked with filter 16 fans, PAH-FN-8 A and B. It is not interlocked with the WG radiation monitors. This is a common misconception. The valve is operated at CP-38.</p> <p>D is incorrect but plausible. VG-V-57 is the WG header isolation to filter 16. It is interlocked with filter 16 fans, PAH-FN-8 A and B. It is not interlocked with the WG radiation monitors. This is a common misconception. The valve is operated at CP-38.</p>		
Technical Reference(s):	OS1252.01, "Process or Effluent High Radiation" Rev 17	
Proposed references to be provided to applicants during examination:	None	
Learning Objective:	SBK LOP L8064I 02, 03	
Question Source:	Bank #	
	Modified Bank#	(Note changes or attach Parent)
	New	x
Question History:		
Question Cognitive Level:	Memory or Fundamental Knowledge	x
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	(5)

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Comments:		

Examination Outline Cross-reference:		Level	RO	SRO
Q52		Tier #	2	
		Group #	1	
		K/A #	076 (SF4S SW) Service Water A3.02 Ability to monitor automatic operation of the SWS, including: Emergency heat loads	
		Importance Rating	3.7	
Proposed Question:				
<p>On an automatic Tower Actuation signal for both 'A' and 'B' trains, what Service Water loads will automatically isolate?</p> <p>A. Emergency Diesel Generator jacket water heat exchangers. B. PAB Fire Protection Booster Pump (FP-P-374). C. PCCW heat exchangers. D. Secondary heat loads.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
<p>D is correct. Since BOTH trains of Service Water have received a TA signal then the turbine building train related SW isolation valves (SW-V-4 and SW-V-5) will have closed. This will isolate SW to the secondary heat loads including SCCW heat exchangers and water box priming pump heat exchangers.</p> <p>A is incorrect but plausible. The Emergency Diesel Generator jacket water heat exchanger does have automatic isolation valves however they are designed to open upon a start of the EDG. The valves are currently maintained open to prevent fouling in the heat exchangers.</p> <p>B is incorrect but plausible. The PAB Fire Protection Booster Pump (FP-P-374) supply is from the SW system within the PAB. It is plausible that the FP booster pump subsystem would be isolated in the event of a TA to prevent potentially pumping down the cooling tower inventory. There is no automatic isolation of this subsystem.</p> <p>C is incorrect but plausible. The PCCW heat exchangers do have automatic isolation valves</p>				

however they are designed to open and prevent manual closure upon a TA signal.			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8037I 13		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 35022
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:		2010 Seabrook NRC Exam	
Question Cognitive Level:	Memory or Fundamental Knowledge	<input checked="" type="checkbox"/>	
	Comprehension or Analysis	<input type="checkbox"/>	
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q53		Tier #		
		Group #		
		K/A #	078 (SF8 IAS) Instrument Air A4.01 Ability to manually operate and/or monitor in the control room: Pressure gauges	
		Importance Rating	3.1	
Proposed Question:				
<p>Plant Conditions:</p> <ul style="list-style-type: none"> SA-SKD-137-A is tagged out for maintenance. SA-SKD-137-B is selected to LEAD. SA-SKD-137-C is selected to LAG. Sullair is available. SA-SKD-137-B trips due to a motor fault. Instrument air pressure is 105 psig and decreasing. <p>With no operator actions, what is the current status of the SA compressors?</p> <p>A. SA-SKD-137-C running; Sullair running. B. SA-SKD-137-C running; Sullair in standby. C. SA-SKD-137-C in standby; Sullair running. D. SA-SKD-137-C in standby; Sullair in standby.</p>				
Proposed Answer:	B.			
Explanation (Optional):				
<p>B is correct. With the lead compressor tripped off, the lag compressor will start and attempt to maintain pressure between 110 and 120 psig. The Sullair auto start setpoint is 100 psig and with the given conditions will not be running, remaining in standby.</p> <p>A is incorrect but plausible. With the lead compressor tripped off, the lag compressor will start and attempt to maintain pressure between 110 and 120 psig. The Sullair auto start setpoint is 100 psig</p>				

<p>and with the given conditions will not be running, remaining in standby.</p> <p>C is incorrect but plausible. B is correct. With the lead compressor tripped off, the lag compressor will start and attempt to maintain pressure between 110 and 120 psig. The Sullair auto start setpoint is 100 psig and with the given conditions will not be running, remaining in standby.</p> <p>D is incorrect but plausible. System pressure has not yet reached the Sullair starting setpoint and it is conceivable that a motor fault on the B compressor would affect the C compressor.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8023I 16		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 35059
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		<input type="checkbox"/>
	Comprehension or Analysis		<input checked="" type="checkbox"/>
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q54	Tier #	2	
	Group #	1	
	K/A #	103 (SF5 CNT) Containment A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation	
	Importance Rating	3.5	
Proposed Question:			
<p>An event occurs that results in repositioning of multiple components on the main control board. The control room operator notes that the following valves have <u>closed</u>:</p> <ul style="list-style-type: none"> • CS-V-168, "Reactor Coolant Pump Seal Water Return Valve". • CS-V-150, "Letdown Line ORC Isolation Valve". • CS-V-145, "Letdown Regen Heat Exchanger Isolation Valve". • CC-V-57, "CC Isolation to Containment". <p>What event has occurred?</p> <p>A. An inadvertent Phase 'A' Isolation signal has occurred. B. Vital 120VAC Instrument Panel 1A has de-energized. C. Instrument Air System pressure is degrading. D. DBA LOCA inside containment.</p>			

Proposed Answer:	D.		
Explanation (Optional):			
<p>D is correct. A DBA LOCA will cause a 'T' and 'P' signal among other signals. 'T' and 'P' are synonymous with a Phase 'A' and 'B' isolation. CS-V-168 and 150 will close on a T signal, CS-V-145 will close once 150 is closed and CC-V-57 will close on the 'P' signal.</p> <p>A is incorrect but plausible. An inadvertent Phase 'A' isolation will account for the first three given valves closing, but will not result in CC-V-57 closing.</p> <p>C is incorrect but plausible. CS-V-150 and CC-V-57 are AOVs that will fail closed on a loss if IA. CS-V-145 will close once 150 is closed. CS-V-168 is a motor operated valve that would not be affected by a loss of instrument air.</p> <p>B is incorrect but plausible. A loss of vital instrument panel 1A would cause a loss of letdown however it would be based on RC-LCV-459 (Train A) closing vice CS-V-150 closing. Additionally, CS-V-168 is a Train B valve that is not affected by a loss of vital instrument panel 1A as that panel is associated with Train A components. The loss of the vital power panel will not result in CC-V-57 closing.</p>			
Technical Reference(s):	OS1205.01, "Inadvertent Phase 'A' Containment Isolation", Rev 17		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1181I 14		
Question Source:	Bank #	x	TEB 35049
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(5)	

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Comments:		

Examination Outline Cross-reference:	Level	RO	SRO																
Q55	Tier #	2																	
	Group #	1																	
	K/A #	103 (SF5 CNT) Containment A3.01 Ability to monitor automatic operation of the containment system, including: Containment isolation																	
	Importance Rating	3.9																	
Proposed Question:																			
<p>An RCS leak resulted in the following conditions:</p> <table border="0"> <thead> <tr> <th><u>TIME</u></th> <th><u>EVENT</u></th> </tr> </thead> <tbody> <tr> <td>0812</td> <td>Manual Reactor Trip.</td> </tr> <tr> <td>0826</td> <td>Pressurizer Pressure 1850 psig and lowering.</td> </tr> <tr> <td>0828</td> <td>Manual Safety Injection.</td> </tr> <tr> <td>0830</td> <td>Pressurizer pressure 1800 psig and lowering</td> </tr> <tr> <td>0907</td> <td>Containment Pressure 4.3 psig and rising.</td> </tr> <tr> <td>0941</td> <td>Containment Pressure 18 psig and rising.</td> </tr> <tr> <td>1003</td> <td>RCS Pressure 220 psig and stable.</td> </tr> </tbody> </table> <p>Assuming NO additional actions were taken, which ONE of the following choices describes the EARLIEST time a Containment Isolation signal was generated?</p> <p>A. 0828 B. 0830 C. 0907 D. 0941</p>				<u>TIME</u>	<u>EVENT</u>	0812	Manual Reactor Trip.	0826	Pressurizer Pressure 1850 psig and lowering.	0828	Manual Safety Injection.	0830	Pressurizer pressure 1800 psig and lowering	0907	Containment Pressure 4.3 psig and rising.	0941	Containment Pressure 18 psig and rising.	1003	RCS Pressure 220 psig and stable.
<u>TIME</u>	<u>EVENT</u>																		
0812	Manual Reactor Trip.																		
0826	Pressurizer Pressure 1850 psig and lowering.																		
0828	Manual Safety Injection.																		
0830	Pressurizer pressure 1800 psig and lowering																		
0907	Containment Pressure 4.3 psig and rising.																		
0941	Containment Pressure 18 psig and rising.																		
1003	RCS Pressure 220 psig and stable.																		
Proposed Answer:	A.																		
Explanation (Optional):																			
A. Correct. The Containment Phase 'A' Isolation ("T" Signal) is actuated via a Safety Injection																			

<p>signal (automatic or manual). The Containment Phase 'B' Isolation ("P" Signal) is actuated via a Containment Building Spray signal (automatic or manual). At time 0828 a manual SI signal was actuated, which would in turn actuate the Containment Phase 'A' Isolation ("T" Signal).</p>			
<p>B. Incorrect but plausible. It is plausible that the student would incorrectly believe that only an automatic SI signal would actuate a Containment Phase 'A' Isolation ("T" Signal). If this were the case, then the student could surmise that the Containment Phase 'A' Isolation ("T" Signal) occurs when the Pressurizer Pressure Low SI setpoint (1800 psig) is reached at 0830.</p>			
<p>C. Incorrect but plausible. It is plausible that the student would incorrectly believe that only an automatic SI signal would actuate a Containment Phase 'A' Isolation ("T" Signal). If this were the case, then the student could surmise that the Containment Phase 'A' Isolation ("T" Signal) occurs when the Containment Pressure Hi-1 setpoint (4.3 psig) is reached at 0907.</p>			
<p>D. Incorrect but plausible. It is plausible that the student would incorrectly surmise that the Containment Phase 'B' Isolation ("P" Signal) was first to occur a) if they misread the conditions in the question stem or b) they incorrectly believe that only an automatic SI signal would actuate a Containment Phase 'A' Isolation ("T" Signal).</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8057I 10		
Question Source:	Bank #	x	
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2015 Seabrook NRC Exam (same K/A) (Question used on one of the two previous NRC exams)		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO										
Q56		Tier #	2											
		Group #	2											
		K/A #	001 (SF1 CRDS) Control Rod Drive K5.64 Knowledge of the following operational implications as they apply to the CRDS: Reason for withdrawing shutdown group: to provide adequate shutdown margin.											
		Importance Rating	3.3											
Proposed Question:														
<p>While performing a reactor startup with manual control rod withdrawal, the operator must verify <u> (1) </u> within 15 minutes of the next control bank withdrawal and the basis of this verification it to ensure <u> (2) </u>.</p> <table style="width: 100%; border: none;"> <tr> <td style="text-align: center; width: 50%;">(1)</td> <td style="text-align: center; width: 50%;">(2)</td> </tr> <tr> <td>A. all shutdown rods are withdrawn</td> <td>adequate shutdown margin</td> </tr> <tr> <td>B. boron concentration at the ECP value</td> <td>adequate shutdown margin</td> </tr> <tr> <td>C. all shutdown rods are withdrawn</td> <td>criticality is achieved at the required rod position</td> </tr> <tr> <td>D. boron concentration at the ECP value</td> <td>criticality is achieved at the required rod position</td> </tr> </table>					(1)	(2)	A. all shutdown rods are withdrawn	adequate shutdown margin	B. boron concentration at the ECP value	adequate shutdown margin	C. all shutdown rods are withdrawn	criticality is achieved at the required rod position	D. boron concentration at the ECP value	criticality is achieved at the required rod position
(1)	(2)													
A. all shutdown rods are withdrawn	adequate shutdown margin													
B. boron concentration at the ECP value	adequate shutdown margin													
C. all shutdown rods are withdrawn	criticality is achieved at the required rod position													
D. boron concentration at the ECP value	criticality is achieved at the required rod position													
Proposed Answer:		A.												
Explanation (Optional):														
<p>A is correct. Per OS1000.07, "Approach to Criticality" within 15 minutes before each 50 step control rod withdrawal verification that all shutdown rods are withdrawn is required. The basis for the shutdown rods being withdrawn is to ensure adequate SDM.</p> <p>B is incorrect but plausible. The boron concentration is calculated by the ECP. The specific boron concentration is established to allow criticality to occur at a desired rod height. Because the rod</p>														

height affects the SDM this is plausible.			
C is incorrect but plausible. Per OS1000.07, "Approach to Criticality" within 15 minutes before each 50 step control rod withdrawal verification that all shutdown rods are withdrawn is required. The shutdown rods being partly inserted would affect the critical rod position but this is not the basis for the verification.			
D is incorrect but plausible. The boron concentration is calculated by the ECP. The specific boron concentration is established to allow criticality to occur at a desired rod height, but this is not required to be verified within 15 minutes.			
Technical Reference(s):		OS1000.07, "Approach to Criticality", Rev 16 steps 4.4.5, 4.4.11.5, 4.4.12.5, etc.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1162I 03		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q57		Tier #	2	
		Group #	2	
		K/A #	011 (SF2 PZR LCS) Pressurizer Level Control K4.06 Knowledge of PZR LCS design feature(s) and/or interlock(s) which provide for the following: Letdown isolation	
		Importance Rating	3.3	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 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. • No operator actions are taken. <p>How do RC-LK-459 and CS-FK-121 respond?</p> <p>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.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
<p>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</p>				

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):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L80271 05		
Question Source:	Bank #	x	TEB 32900
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2013 Seabrook NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO										
Q58		Tier #	2											
		Group #	2											
		K/A #	028 (SF5 HRPS) Hydrogen Recombiner and Purge Control A1.01 Ability to predict and/or monitor changes in parameter (to prevent exceeding design limits) associated with operating the HRPS controls including: Hydrogen concentration											
		Importance Rating	3.4											
Proposed Question:														
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Large LOCA. • Several safety systems have failed. • The crew is processing FR-C.1, "Response to Inadequate Core Cooling". • The hydrogen analyzers have been placed in service. • Hydrogen concentration is 2.5%. <p>What action (1) is required based upon this hydrogen concentration and why (2)?</p> <table style="width: 100%; border: none;"> <tr> <td style="text-align: center; width: 50%;">(1)</td> <td style="text-align: center; width: 50%;">(2)</td> </tr> <tr> <td>A. Start the hydrogen recombiners.</td> <td>Any hydrogen burn will not produce a significant pressure rise.</td> </tr> <tr> <td>B. Start the hydrogen recombiners.</td> <td>Concentration is above the flammability limit.</td> </tr> <tr> <td>C. Do not start the hydrogen recombiners.</td> <td>Any hydrogen burn will not produce a significant pressure rise.</td> </tr> <tr> <td>D. Do not start the hydrogen recombiners.</td> <td>Concentration is above the flammability limit.</td> </tr> </table>					(1)	(2)	A. Start the hydrogen recombiners.	Any hydrogen burn will not produce a significant pressure rise.	B. Start the hydrogen recombiners.	Concentration is above the flammability limit.	C. Do not start the hydrogen recombiners.	Any hydrogen burn will not produce a significant pressure rise.	D. Do not start the hydrogen recombiners.	Concentration is above the flammability limit.
(1)	(2)													
A. Start the hydrogen recombiners.	Any hydrogen burn will not produce a significant pressure rise.													
B. Start the hydrogen recombiners.	Concentration is above the flammability limit.													
C. Do not start the hydrogen recombiners.	Any hydrogen burn will not produce a significant pressure rise.													
D. Do not start the hydrogen recombiners.	Concentration is above the flammability limit.													
Proposed Answer:	A.													
Explanation (Optional):														

A is correct. Step 8 of FR-C.1 checks hydrogen concentration. If hydrogen concentration is between 0.5 and 4.0%, the hydrogen recombiners will be placed in service. The recombiners will be effective in reducing hydrogen concentration. Any hydrogen burn will not produce a significant pressure rise.

B is incorrect but plausible. The hydrogen recombiners will not be placed in service if concentration is above the flammability limit.

C is incorrect but plausible. If hydrogen concentration is less than 0.5%, a flammable situation is not imminent and the recombiners will not be placed in service.

D is incorrect but plausible. If hydrogen concentration is greater than 4% the recombiners will not be placed in service as concentration is above the flammability limit.

Technical Reference(s):		FR-C.1' Response to Inadequate Core Cooling" Rev 28	
		Background document for FR-C.1, Rev 3	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1227I 02		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO															
Q59	Tier #	2																
	Group #	2																
	K/A #	041 (SF4S SDS) Steam Dump/Turbine Bypass Control A4.02 Ability to manually operate and/or monitor in the control room: Cooldown valves																
	Importance Rating	2.7																
Proposed Question:																		
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Startup is in progress per OS1000.02, "Plant Startup from Hot Standby to Minimum Load". • Power is 3% at MOL. • Tavg is 559 °F. • Steam dumps are in Steam Pressure mode with MS-PK-507 in AUTO. • Preparations are being made to enter Mode 1. • MS-PK-507 OUTPUT fails to 100%. • All plant systems respond as designed. • Assume no operator action. <p>How do the Steam dumps and reactor power initially respond?</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 30%;"></td> <td style="text-align: center;"><u>Steam Dumps</u></td> <td style="text-align: center;"><u>Reactor Power</u></td> </tr> <tr> <td>A.</td> <td>Close</td> <td>Increases</td> </tr> <tr> <td>B.</td> <td>Open</td> <td>Increases</td> </tr> <tr> <td>C.</td> <td>Close</td> <td>Decreases</td> </tr> <tr> <td>D.</td> <td>Open</td> <td>Decreases</td> </tr> </table>					<u>Steam Dumps</u>	<u>Reactor Power</u>	A.	Close	Increases	B.	Open	Increases	C.	Close	Decreases	D.	Open	Decreases
	<u>Steam Dumps</u>	<u>Reactor Power</u>																
A.	Close	Increases																
B.	Open	Increases																
C.	Close	Decreases																
D.	Open	Decreases																

Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. Given the plant conditions in the stem of the question, PK-507 output failing high will cause the steam dumps to open. The increased steam flow will cause Tavg to decrease and reactor power will increase.</p> <p>A is incorrect but plausible. The student could mistake that PK-507 output failing high would cause the steam dumps to close. Additionally, reactor power could increase if the MTC is slightly positive. However, MTC is negative since it is MOL.</p> <p>C is incorrect but plausible. The student could mistake that PK-507 output failing high would cause the steam dumps to close.</p> <p>D is incorrect but plausible. Reactor power could decrease if the MTC is slightly positive. However, MTC is negative since it is MOL.</p>			
Technical Reference(s):	N/A		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L8047I 15		
Question Source:	Bank #	x	TEB 31637
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41	(7)	

2020 Seabrook Station NRC Written Exam
ES-401-5 Written Examination Question Worksheet

	55.43	
Comments:		

Examination Outline Cross-reference:		Level	RO	SRO
Q60		Tier #	2	
		Group #	2	
		K/A #	045 (SF 4S MTG) Main Turbine Generator A2.17 Ability to (a) predict the impacts of the following malfunctions or operation on the MT/G system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Malfunction of electrohydraulic control	
		Importance Rating	2.7	
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • Control Valve 4 is slowly closing due a failure in the EHC system. • Control Rods are in AUTO. • The crew is performing OS1231.03, "Turbine Runback/Setback". • During the transient SG pressures are observed to be 1155 psig and increasing. <p>What action will the crew take?</p> <p>A. Once power is less than P-9, trip the turbine. B. Manually open steam dumps. C. Manually insert control rods. D. Trip the reactor.</p>				
Proposed Answer:	D.			

Explanation (Optional):			
<p>D is correct. If SG pressures are not less than 1150 psig, OS1231.03, "Turbine Runback/Setback" will direct the crew to trip the reactor. This is done to prevent actuation of the SG safety valves.</p> <p>A is incorrect but plausible. With plant power automatically lowering, it is plausible that a turbine trip would be required due to the high SG pressures once power is <P-9.</p> <p>B is incorrect but plausible. OS1231.03 checks for proper steam dump operation. If steam dumps are malfunctioning, the crew will manually operate steam dumps. It is plausible that the reason for the high SG pressures is a malfunction of the steam dumps.</p> <p>C is incorrect but plausible. With control rods in AUTO as given, the automatic load reduction due to the failure of the EHC system will cause control rods to insert. The student must be able to predict this plant response. Only if control rods were failing to insert would rods be taken to manual. Rods are not taken to manual in response to high SG pressures.</p>			
Technical Reference(s):		OS1231.03, "Turbine Runback/Setback". Rev 23	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1183I 03		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(5)	
	55.43		
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q61		Tier #	2	
		Group #	2	
		K/A #	055 (SF4S CARS) Condenser Air Removal K3.01 Knowledge of the effect that a loss or malfunction of the CARS will have on the following: Main condenser	
		Importance Rating	2.5	
Proposed Question:				
<p>The following plant conditions exist:</p> <ul style="list-style-type: none"> • 35% power. • The running mechanical vacuum pump trips. • Condenser Low Vacuum Hardwire Alarm has just been acknowledged. • The crew notes condenser back-pressure continues to degrade. <p>What is the NEXT expected plant response if NO operator action is taken?</p> <p>A. The standby mechanical vacuum pump starts. B. The turbine will trip, resulting in a reactor trip. C. The turbine will trip and the reactor will not trip. D. The main feedwater pumps will trip, resulting in a turbine trip.</p>				
Proposed Answer:	C.			
Explanation (Optional):				
<p>C is correct. The low condenser vacuum hardwire alarm actuates at 24.9 "HgV. If vacuum continues to degrade the next event that will occur is a turbine trip. Because power is below P-9 (45%) the reactor will not trip.</p> <p>A is incorrect but plausible. The standby mechanical vacuum pump starts at 26 "HgV. It should already be running for the given plant conditions.</p>				

B is incorrect but plausible. The turbine will trip however, because power is less than P-9 the reactor will not trip.

D is incorrect but plausible. The main feed water pumps will trip on low vacuum at 18.5 "HgV. This is not the next action in accordance with the stem though. This will occur after the turbine trip if vacuum continues to degrade.

- Normal Vacuum >27.2 "HgV
- MPCS Alarm <27.0 "HgV
- Mech Vac Pump Auto Start <26.0 "HgV
- Steam Dump Block <25.0 "HgV
- Hardwire Alarm <24.9 "HgV
- Main Turbine Trip <22.4 "HgV
- SGFP Turbine Trip <18.5 "HgV

Technical Reference(s):		N/A		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8042I 02			
Question Source:	Bank #	x	TEB 30785	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41	(7)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q62	Tier #	2	
	Group #	2	
	K/A #	056 (SF1 RPIS) Rod Position Indication System 2.2.37 Ability to determine operability and/or availability of safety related equipment.	
	Importance Rating	3.6	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 88% power. • Control Bank 'D' Group Demand Counters indicate 218 steps. • Failure of DRPI Data 'B'. • Accuracy Mode Selector Switch has been placed to 'A' Only. • The General Warning lights are flashing for all rods. <p>How is the operability of the Digital Rod Position Indication system affected?</p> <p>A. Operable and capable of determining rod position within ± 6 steps.</p> <p>B. Operable and capable of determining rod position within ± 12 steps.</p> <p>C. Inoperable. Determine that shutdown margin requirement is satisfied within 1 hour and be in hot standby within 6 hours.</p> <p>D. Inoperable. Within 1-hour action shall be initiated to place the unit in a MODE in which the specification does not apply.</p>			
Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. Each set of DRPI coils, A and B are placed 12 steps apart. DRPI is operable having only data A or data B coils, and likewise is operable when placing the Accuracy Mode Selector switch in 'A Only'. TS 3.1.3.2 requires DRPI be capable of determining control rod positions to within ± 12 steps.</p>			

<p>A is incorrect but plausible. It is conceivable that because of the arrangement of DRPI coils, that only data A would resolve rod position to within ± 6 steps.</p> <p>C is incorrect but plausible. TS 3.1.3.1 requires this action for inoperable control rods. It is a common misconception that inoperable DRPI implies that control rods are also inoperable.</p> <p>D is incorrect but plausible. It is conceivable that the loss of DRPI redundancy causes all control rods to be inoperable. This is not a defined TS condition and has no applicable action, thus TS 3.0.3 would apply.</p>			
Technical Reference(s):		Tech Spec 3.1.3.2	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8032I 08, 11		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 32841
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		<input type="checkbox"/>
	Comprehension or Analysis		<input checked="" type="checkbox"/>
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO										
Q63	Tier #	2											
	Group #	2											
	K/A #	072 (SF7 ARM) Area Radiation Monitoring K1.04 Knowledge of the physical connections and/or cause effect relationships between the ARM system and the following systems: Control room ventilation.											
	Importance Rating	3.3											
Proposed Question:													
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 'A' CBA is in normal alignment with CBA-FN-27A and CBA-FN-14A running. • RM-6506B, "CONTROL BLDG EAST AIR INTK RAD MONITOR" goes into high alarm from a valid high radiation signal. <p>Following automatic system response:</p> <p>CBA-FN-27A, "TRAIN A CONTROL ROOM MAKE UP AIR FAN 27A" will be <u> (1) </u>.</p> <p>CBA-FN-14A, "CONTROL ROOM AIR CONDITIONING FAN" will be <u> (2) </u>.</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. running</td> <td>stopped</td> </tr> <tr> <td>B. running</td> <td>running</td> </tr> <tr> <td>C. stopped</td> <td>stopped</td> </tr> <tr> <td>D. stopped</td> <td>running</td> </tr> </table>				(1)	(2)	A. running	stopped	B. running	running	C. stopped	stopped	D. stopped	running
(1)	(2)												
A. running	stopped												
B. running	running												
C. stopped	stopped												
D. stopped	running												
Proposed Answer:	D.												
Explanation (Optional):													
D is correct. On a single train actuation of control room filter recirculation (CRFRM), the respective supply fan will trip, in this case that is FN-27B. The supply fan dampers are cross trained to ensure that both fans will trip on a single train actuation. The damper closing will cause the fan to trip.													

<p>Thus FN-27A will be stopped from the 'B' train CRFRM actuation. The 14 fans are the air conditioning recirculation fans supplying air from the mechanical room to the control room complex. These fans are unaffected from a CRFRM actuation and will remain running. This is a common misconception.</p> <p>Distractors A, B and C are incorrect but plausible. In part (2) it is a common misconception that a CRFRM signal will stop the control room air conditioning fans. This is not the case. In part (1) it is also a common misconception that FN-27 will continue to run after a CRFRM signal. A CRFRM signal will stop FN-27 A/B and start FN-16 A/B. There exists much confusion between FN-27 and 16 in non-competent operators.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8039I 05		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(2)-(9)	
	55.43		
<p>K/A match justification: Rad monitors in this question are airborne radiation monitors. Seabrook has area, airborne and process radiation monitors. NUREG 1122 divides radiation monitors into process and area radiation monitors only. Seabrook's airborne radiation monitors are effectively area radiation monitors.</p>			

Examination Outline Cross-reference:		Level	RO	SRO										
Q64		Tier #	2											
		Group #	2											
		K/A #	033 (SF8 SFPCS) Spent Fuel Pool Cooling System A3.01 Ability to monitor automatic operation of the Spent Fuel Pool Cooling System including: Temperature control valves											
		Importance Rating	2.5											
Proposed Question:														
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power • Spent Fuel cooling pump SF-P-10A is in service. • Large break LOCA inside containment. <p>How is Spent Fuel cooling impacted (1) and what actions required to restore normal alignment (2)?</p> <table style="width: 100%; border: none;"> <tr> <td style="text-align: center; width: 50%;">(1)</td> <td style="text-align: center; width: 50%;">(2)</td> </tr> <tr> <td>A. 'S' signal has tripped SF-P-10A.</td> <td>Reset 'S' signal and restart the pump locally.</td> </tr> <tr> <td>B. 'S' signal has tripped SF-P-10A.</td> <td>Reset 'S' signal and restart the pump at the MCB.</td> </tr> <tr> <td>C. 'T' signal has isolated cooling to the SF hx.</td> <td>Reset the 'T' signal and reopen valve locally.</td> </tr> <tr> <td>D. 'T' signal has isolated cooling to the SF hx.</td> <td>Reset the 'T' signal and reopen valve at the MCB.</td> </tr> </table>					(1)	(2)	A. 'S' signal has tripped SF-P-10A.	Reset 'S' signal and restart the pump locally.	B. 'S' signal has tripped SF-P-10A.	Reset 'S' signal and restart the pump at the MCB.	C. 'T' signal has isolated cooling to the SF hx.	Reset the 'T' signal and reopen valve locally.	D. 'T' signal has isolated cooling to the SF hx.	Reset the 'T' signal and reopen valve at the MCB.
(1)	(2)													
A. 'S' signal has tripped SF-P-10A.	Reset 'S' signal and restart the pump locally.													
B. 'S' signal has tripped SF-P-10A.	Reset 'S' signal and restart the pump at the MCB.													
C. 'T' signal has isolated cooling to the SF hx.	Reset the 'T' signal and reopen valve locally.													
D. 'T' signal has isolated cooling to the SF hx.	Reset the 'T' signal and reopen valve at the MCB.													
Proposed Answer:	D.													
Explanation (Optional):														

<p>D is correct. The LOCA will cause an 'S' and 'T' signal to actuate. A 'T' signal will cause PCCW valves to SFP heat exchangers to close. To restore SFP cooling the 'T' signal must be reset and the valve reopened from the MCB.</p> <p>A and B are incorrect but plausible. An 'S' signal will cause many components to change status.</p> <p>C is incorrect but plausible. The SFP heat exchangers are placed in service by locally aligning SFP water. CC is aligned at the MCB.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8061I 05		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
<p>K/A match justification: PCCW valves CC-V-32 and 445 are the CC supply valves to the SFP heat exchangers in 'A' and 'B' train, respectively. These valves are closed upon a 'T' signal. The valves are either full open or full closed, effectively acting as temperature control valves. Normally the valves are full open aligning full cooling to the SFP heat exchangers. After a 'T' signal that cooling is lost.</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q65	Tier #	2	
	Group #	2	
	K/A #	086 Fire Protection K6.04 Knowledge of the effect of a loss or malfunction on the Fire Protection System following will have on the: Fire, smoke, and heat detectors.	
	Importance Rating	2.6	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • New fuel is being transferred from the new fuel storage area into the spent fuel pool. • 'A' train FAH is in the fuel handling mode. • A carbon monoxide instrument in zone #FAH-F-41 on MM-CP-517 is found to be nonfunctional. <p>What action is required (if any) in order to continue use of FAH-F-41, "Train 'A' FSB Cleanup Filter"?</p> <p>(reference provided)</p> <p>A. Within 1 hour establish a watch to monitor MM-CP-517. B. Within 1 hour establish a continuous fire watch. C. Within 1 hour establish an hourly fire patrol. D. None. Use the redundant instrument.</p>			
Proposed Answer:	B.		
Explanation (Optional):			
<p>B is correct. TR12-3.3.3.7 action e. gives requirements for nonfunctional CO monitoring instrumentation. Sheet 10 of 10 lists zone FAH-F-41 with 2 CO instruments. In order for the filter unit to remain in service, action e.1 applies to establish a continuous fire watch within 1 hour.</p> <p>A is incorrect but plausible. TR12-3.3.3.7 action f. is to establish a watch to monitor a non-</p>			

<p>communicating fire panel within 1 hour. This does not apply for a single non-functional CO detector.</p> <p>C is incorrect but plausible. TR12-3.3.3.7 action e. gives requirements for nonfunctional CO monitoring instrumentation. If the fan is secured it is permissible to establish an hourly fire patrol however, the stem of the question asks what action is required in order to continue use of the fan.</p> <p>D is incorrect but plausible. It is possible that the student interprets the loss of 1 instrument to not require any action as there is a redundant instrument available.</p>			
Technical Reference(s):		TR12-3.3.3.7 Rev 137	
Proposed references to be provided to applicants during examination:			TR12-3.3.3.7 Rev 137 pages 2-12.1 through 2-12.10
Learning Objective:	SBK LOP L8089I 14		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q66	Tier #	3	
	Group #		
	K/A #	Conduct of Operations 2.1.15 Knowledge of administrative requirements for temporary management directives such as standing orders, night orders, Operations memos, etc.	
	Importance Rating	2.7	
Proposed Question:			
<p>Which of the following problems would be addressed as a Standing Operation Order in accordance with the OPMM?</p> <p>A. Only conditions that do NOT require a 50.59 evaluation.</p> <p>B. A required valve position interlock that allows turbine shell and chest warming is nonfunctional, so a jumper must be installed.</p> <p>C. The turbine power set points of the C-20 AMSAC permissive must be temporarily raised to allow digital EHC panel work.</p> <p>D. Direct use of alternate indication to verify containment isolation valve position instead of the critical safety function status tree.</p>			
Proposed Answer:	D.		
Explanation (Optional):			
<p>D is correct. This direction does not violate OPMM chapter 6 for what a SOO can or cannot be used for.</p> <p>A is incorrect but plausible. 50.59 screenings are performed on all SOOs and a 50.59 evaluation is done in accordance with the NARC if required.</p> <p>B is incorrect but plausible. SOOs cannot circumvent the TMOD/TALT process.</p> <p>C is incorrect but plausible. SOOs cannot be used to bypass a SORC approved procedure. This setpoint is a SORC approved setpoint that is described in the UFSAR.</p>			

2020 Seabrook Station NRC Written Exam
 ES-401-5 Written Examination Question Worksheet

Technical Reference(s):		OPMM Rev 109. page 6-1.1		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L1305I 08			
Question Source:	Bank #	x	12023	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	2009 Seabrook NRC exam			
Question Cognitive Level:	Memory or Fundamental Knowledge		x	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	(10)		
	55.43			
Comments:				

Examination Outline Cross-reference:		Level	RO	SRO																									
Q67		Tier #	3																										
		Group #																											
		K/A #	Conduct of Operations 2.1.18 Ability to make accurate, clear and concise logs, records, status boards and reports.																										
		Importance Rating	3.6																										
Proposed Question:																													
<p>The PSO identifies an event that occurred during the previous shift that should be entered into the Narrative Log.</p> <p>In accordance with OP-AA-100-1000, "Conduct of Operations", which of the following indicates the correct individual (1) and necessary steps (2) to make this entry?</p> <table border="0" style="width: 100%;"> <tr> <td style="width: 30%;"></td> <td style="text-align: center;">(1)</td> <td style="text-align: center;">(2)</td> <td colspan="2"></td> </tr> <tr> <td>A. Any Watch Stander</td> <td></td> <td>Record a description of the event, use the actual time the event occurred ONLY.</td> <td colspan="2"></td> </tr> <tr> <td>B. Any Watch Stander</td> <td></td> <td>Record a description of the event, use the actual time the event occurred, and designate the event with Late Entry.</td> <td colspan="2"></td> </tr> <tr> <td>C. The SM ONLY</td> <td></td> <td>Record a description of the event, use the actual time the event occurred ONLY.</td> <td colspan="2"></td> </tr> <tr> <td>D. The SM ONLY</td> <td></td> <td>Record a description of the event, use the actual time the event occurred, and designate the event with Late Entry.</td> <td colspan="2"></td> </tr> </table>						(1)	(2)			A. Any Watch Stander		Record a description of the event, use the actual time the event occurred ONLY.			B. Any Watch Stander		Record a description of the event, use the actual time the event occurred, and designate the event with Late Entry.			C. The SM ONLY		Record a description of the event, use the actual time the event occurred ONLY.			D. The SM ONLY		Record a description of the event, use the actual time the event occurred, and designate the event with Late Entry.		
	(1)	(2)																											
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Proposed Answer:	B.																												
Explanation (Optional):																													
B is correct. Per OP-AA-100-1000, "Conduct of Operations" and ODI-28 "Proper Journal and Log Maintenance", there are no restrictions on who may make a late entry, only that the actual time the event occurred and a designation of "Late Entry".																													

<p>A is incorrect but plausible. Part (1) is correct. If the student is not aware of the administrative requirements for late entries and cannot demonstrate their ability to make accurate logs, this would be a plausible answer.</p> <p>C and D are incorrect but plausible. OP-AA-100-1000 section 3.1.6 gives operations senior management (SM) the responsibility to review the control room logs daily. It is plausible that the SM is the only individual with the authority to make late entries.</p>			
<p>Technical Reference(s):</p>		<p>OP-AA-100-1000, "Conduct of Operations" Rev 31</p> <p>ODI-28 Rev 31</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1305I 10</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>x</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:		Level	RO	SRO										
Q68		Tier #	3											
		Group #												
		K/A #	Equipment Control 2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.											
		Importance Rating	3.9											
Proposed Question:														
<p>A Primary to Secondary leak of 0.15 gpm exists. The associated leakage Tech Spec LCO is ____(1)____ and this is type of leakage is defined as ____(2)____.</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. Met</td> <td>Unidentified</td> </tr> <tr> <td>B. Met</td> <td>Identified</td> </tr> <tr> <td>C. Exceeded</td> <td>Unidentified</td> </tr> <tr> <td>D. Exceeded</td> <td>Identified</td> </tr> </table>					(1)	(2)	A. Met	Unidentified	B. Met	Identified	C. Exceeded	Unidentified	D. Exceeded	Identified
(1)	(2)													
A. Met	Unidentified													
B. Met	Identified													
C. Exceeded	Unidentified													
D. Exceeded	Identified													
Proposed Answer:		D.												
Explanation (Optional):														
<p>D is correct. Primary to Secondary leakage is defined as Identified leakage by Tech Specs. The limit on primary to secondary leakage is 150 gpd. The given leak rate of 0.15 gpm equates to 216 gpd which is in excess of the allowed limit. The LCO is exceeded.</p> <p>A and C are incorrect but plausible. The student may believe that because the leakage is through a fission product boundary (SG U tubes) it is defined as Unidentified leakage. This is incorrect. TS defines primary to secondary leakage as Identified leakage. The leakage given is equivalent to 216 gpd which exceeded the allowable limit.</p>														

B is incorrect but plausible. Primary to Secondary leakage is defined as Identified leakage by Tech Specs. The limit on primary to secondary leakage is 150 gpd. The given leak rate of 0.15 gpm equates to 216 gpd which is in excess of the allowed limit. The LCO is exceeded.			
Technical Reference(s):		Technical Specifications Rev 141, TS 3.4.6.2	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8010I 10		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41	(7), (10)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q69	Tier #	3	
	Group #		
	K/A #	Equipment Control 2.2.35 Ability to determine Technical Specification Mode of Operation.	
	Importance Rating	3.6	
Proposed Question:			
<p>In accordance with OS1000.07, "Approach to Criticality" when is MODE 2 entered?</p> <p>A. When the reactor is declared critical.</p> <p>B. When the reactor trip breakers are closed.</p> <p>C. When the operators commence control bank withdrawal.</p> <p>D. When the operators commence shutdown bank withdrawal.</p>			
Proposed Answer:	A.		
Explanation (Optional):			
<p>A is correct. From OS1000.07, "Approach to Criticality", "MODE 2 is declared when the reactor is declared critical. MODE 2 Tech. Spec. will be met prior to pulling rods to criticality and MODE 3 Tech. Spec. will be met until critical (i.e., DRPI Tech. Spec.). During an approach to criticality, an adequate shutdown margin is determined by the ECP and Rod Position."</p> <p>B is incorrect but plausible. There are numerous Tech Spec action statements that when violated require the reactor trip breakers to be opened. It is conceivable that closing the trip breakers were an entry into mode 2.</p> <p>C is incorrect but plausible. Withdrawing control banks during the approach to criticality adds positive reactivity and is conceivable that this is the criteria for entry into mode 2.</p> <p>D is incorrect but plausible. Withdrawing shutdown banks during the approach to criticality adds positive reactivity and is conceivable that this is the criteria for entry into mode 2.</p>			

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Technical Reference(s):		OS1000.07, "Approach to Criticality" Rev 16		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L8010I 03			
Question Source:	Bank #	x	TEB 15682	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:				
Question Cognitive Level:	Memory or Fundamental Knowledge		x	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	(7), (10)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q70	Tier #	3	
	Group #		
	K/A #	Radiation Control 2.3.11 Ability to control radiation releases.	
	Importance Rating	3.8	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Containment building pressure is being reduced in accordance with OS1023.69, "Containment Online Purge (COP) System Operation". • COP exhaust containment isolation valves COP-V-3 and COP-V-4 have been opened. • The crew is establishing COP flow through COP-V-8, "COP Exhaust Throttle Valve Coarse Control" • RM-6527A-1 and 6527A-2, "Train 'A' COP Rad Monitor" go into HIGH ALARM. • All systems function as designed. <p>Which of the following describes how the control room crew will control the radiological release?</p> <p>A. Control room operators must ensure COP-V-4 automatically closes to stop the release.</p> <p>B. Control room operators must ensure COP-V-3 and COP-V-8 automatically close to stop the release.</p> <p>C. Control room Operators must ensure COP-V-4 and COP-V-8 automatically close to stop the release.</p> <p>D. Control room operators must manually close COP-V-3 and COP-V-4 since no automatic actions will occur.</p>			
Proposed Answer:	A.		
Explanation (Optional):			
<p>A is correct. COP-V-3 and 4 receive an automatic CVI signal to close when high radiation is sensed. COP Valves 1 & 4 receive a CVI signal from Train 'A'. COP Valves 2 & 3 receive a CVI signal from Train 'B'.</p> <p>B is incorrect but plausible. COP V-3 does receive a CVI signal, however it is from Train B</p>			

<p>radiation monitors. It is a common operator misconception that the COP exhaust throttle valves also receive a CVI signal but this is incorrect (COP-V-8 will not close automatically).</p> <p>C is incorrect but plausible. COP-V-4 is a Train 'A' valve and will receive a CVI signal to close. It is a common operator misconception that the COP exhaust throttle valves also receive a CVI signal but this is incorrect (COP-V-8 will not close automatically).</p> <p>D is incorrect but plausible. Both COP-V- 3 and 4 receive a CVI signal to close, however COP-V-3 receives it's signal from Train 'B'.</p>			
Technical Reference(s):		N/A	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8059I 06		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 35043
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:	2010 Seabrook NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge		<input type="checkbox"/>
	Comprehension or Analysis		<input checked="" type="checkbox"/>
10 CFR Part 55 Content:	55.41	(10)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO										
Q71	Tier #	3											
	Group #												
	K/A #	Radiation Control 2.3.12 Knowledge of radiological safety principles pertaining to licensed operator duties.											
	Importance Rating	3.2											
Proposed Question:													
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Refueling outage in progress. • Incore Instruments have been withdrawn. • Core off-load has commenced. • The crew has noted a 1 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. <p>What is the concern (1), and who can authorize under-vessel access, if at all (2)?</p> <table style="width: 100%; border: none;"> <thead> <tr> <th style="width: 50%; text-align: center;">(1)</th> <th style="width: 50%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A. The highly irradiated Incore Instruments are withdrawn into the area under the reactor vessel.</td> <td>RP Manager must specify conditions and authorize access.</td> </tr> <tr> <td>B. During fuel transfer, under-vessel access is restricted due to rapidly changing dose rates.</td> <td>RP Manager must specify conditions and authorize access.</td> </tr> <tr> <td>C. The highly irradiated Incore Instruments are withdrawn into the area under the reactor vessel.</td> <td>Access cannot be authorized by anyone.</td> </tr> <tr> <td>D. During fuel transfer, under-vessel access is restricted due to rapidly changing dose rates.</td> <td>Access cannot be authorized by anyone.</td> </tr> </tbody> </table>				(1)	(2)	A. The highly irradiated Incore Instruments are withdrawn into the area under the reactor vessel.	RP Manager must specify conditions and authorize access.	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 Incore Instruments are withdrawn into the area under the reactor vessel.	Access cannot be authorized by anyone.	D. During fuel transfer, under-vessel access is restricted due to rapidly changing dose rates.	Access cannot be authorized by anyone.
(1)	(2)												
A. The highly irradiated Incore Instruments are withdrawn into the area under the reactor vessel.	RP Manager must specify conditions and authorize access.												
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C. The highly irradiated Incore Instruments are withdrawn into the area under the reactor vessel.	Access cannot be authorized by anyone.												
D. During fuel transfer, under-vessel access is restricted due to rapidly changing dose rates.	Access cannot be authorized by anyone.												

Proposed Answer:				C.	
Explanation (Optional):					
<p>C 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"</p> <p>A is incorrect but plausible. 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". No one can authorize access.</p> <p>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.</p> <p>D is incorrect but 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.</p>					
Technical Reference(s):			SSRP RP-9.2 Figure 5.1 Rev 14		
Proposed references to be provided to applicants during examination:					None
Learning Objective:	SBK LOP L1307I 01				
Question Source:	Bank #	x	TEB 31453		
	Modified Bank#			(Note changes or attach Parent)	
	New				
Question History:	2013 Seabrook NRC Exam				
Question Cognitive Level:	Memory or Fundamental Knowledge	x			
	Comprehension or Analysis				

10 CFR Part 55 Content:	55.41	(12)
	55.43	
<p>Comments:</p> <p>Question altered from original. Turned into 2x2 to eliminate distracters referring to movable fission chambers.</p> <p>Original question:</p> <p>Plant conditions:</p> <ul style="list-style-type: none"> • 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. <p>What is the concern, and who can authorize under-vessel access, if at all?</p> <p>A. The highly irradiated Incore Instrument thimbles are withdrawn into the area under the reactor vessel. Access cannot be authorized by anyone.</p> <p>B. During fuel transfer, under-vessel access is restricted due to rapidly changing dose rates. RP Manager must specify conditions and authorize access.</p> <p>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.</p> <p>D. Due to the presence of a highly irradiated detached Movable Fission Chamber in Incore Instrument thimble #36 a survey is required. RP Manager must specify conditions and authorize access.</p>		

Examination Outline Cross-reference:	Level	RO	SRO
Q72	Tier #	3	
	Group #		
	K/A #	Radiation Control 2.3.5 Ability to use radiation monitoring systems.	
	Importance Rating	2.9	
Proposed Question:			
<p>From where is a Radiation Monitor Source Check performed?</p> <p>A. Locally at the Radiation Monitor. B. Locally at the RM-80 unit. C. CP-295. D. MPCS.</p>			
Proposed Answer:	C.		
Explanation (Optional):			
<p>C is correct. Per Figure 14 of OS1000.10, rad monitor source checks are performed at CP-295.</p> <p>A is incorrect but plausible. During testing of rad monitors during normal evolutions, rad monitor functional checks are performed. This often involves local verification of equipment response, but does not involve a source check.</p> <p>B is incorrect but plausible. In the event of a failure of communication between a radiation monitor and CP-295, local checks are made on RM-80 units. This does not involve a source check.</p> <p>D is incorrect but plausible. Operators routinely interface with rad monitors on the MPCS. It is plausible that a source check be performed from this location.</p>			
Technical Reference(s):	OS1000.10, "Operation at Power" Figure 14 Rev 44.		
Proposed references to be provided to applicants during examination:	None		

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 ES-401-5 Written Examination Question Worksheet

Learning Objective:	SBK LOP L8059I 08, 09		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(11), (12)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q73	Tier #	3	
	Group #		
	K/A #	Emergency Procedures/Plan 2.4.1 Knowledge of EOP entry conditions and immediate action steps.	
	Importance Rating	4.6	
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • All systems are aligned normally. • Control rods are in MANUAL. • The main turbine has tripped due to high bearing vibrations. • A valid reactor trip signal is received and the reactor did NOT automatically trip. • The Control Room Operator could not manually trip the reactor from the Main Control Board. • The crew has entered FR-S.1, "Response to Nuclear Power Generation/ATWS." <p>What is the <u>FIRST</u> action that should be taken in order to insert negative reactivity into the core?</p> <p>A. Close the Main Steam Isolation Valves and allow the RCS to heat up. B. Align Charging Pump suction to the RWST and isolate suction from the VCT. C. Verify control rods are being inserted in auto OR manually insert control rods. D. Start at least one Boric Acid Pump and OPEN CS-V-426, Emergency Borate Valve.</p>			
Proposed Answer:	C.		
Explanation (Optional):			
C is correct. The response not obtained action for the first step in FR-S.1 (immediate action step) directs a manual trip of the reactor. If the reactor will not trip manually then the step directs the operator to verify that control rods are being inserted in auto OR manually insert control rods.			

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

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

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

Technical Reference(s):		FR-S.1, "Response to Nuclear Power Generation/ATWS" Rev 30		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L1200I 01, 02			
Question Source:	Bank #	x	TEB 35009	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:	2010 Seabrook NRC Exam			
Question Cognitive Level:	Memory or Fundamental Knowledge		x	
	Comprehension or Analysis			
10 CFR Part 55 Content:	55.41	(10)		
	55.43			
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q74	Tier #	3	
	Group #		
	K/A #	Emergency Procedures/Plan 2.4.11 Knowledge of abnormal condition procedures	
	Importance Rating	4.0	
Proposed Question:			
<p>The PSO is going to perform a skill of the operator task before the crew enters an AOP.</p> <p>In accordance with OP 9.2, "Transient Response Procedure User's Guide" which of the following lists two criteria that must be met in order to permit performance of the skill of the operator task?</p> <p>A. A component is deviating from its design function state; There is not sufficient time to obtain concurrence from the supervisor.</p> <p>B. A peer check must be performed; There is not sufficient time to obtain concurrence from the supervisor.</p> <p>C. A component is deviating from its design function state; The task is simple and is a routine activity.</p> <p>D. A peer check must be performed; The task is simple and is a routine activity.</p>			
Proposed Answer:	C.		
Explanation (Optional):			

C is correct. A, B and D are incorrect but plausible.

Per OP9.2 the criteria that are required for a “Skill of the Operator Task” are as follows:

- a. A system or component is deviating from its design function state or anticipated that the system or component will deviate from its design function state.
- b. The task is simple and is considered a routine activity based on operational experience or training.
- c. Written instruction does not exist or written instruction is not immediately available.
- d. The operator obtains concurrence from his/her supervisor prior to performing task.
- e. As time allows, any written instruction should be used as follow up to verify task completion.

Distracters A, B and D are a combination of a correct criteria and an incorrect but plausible criteria.

Technical Reference(s):		OP 9.2, “Transient Response Procedure User’s Guide” Rev 19	
Proposed references to be provided to applicants during examination:			None
Learning Objective:			
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		x
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41	(10)	
	55.43		
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO										
Q75	Tier #	3											
	Group #												
	K/A #	Emergency Procedures/Plan 2.4.46 Ability to verify that the alarms are consistent with the plant conditions.											
	Importance Rating	4.2											
Proposed Question:													
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A large LOCA has occurred. • SI has been reset. • RWST Level is 120,000 gallons and lowering. • Containment recirculation sump level is 3 feet and rising. • All equipment functions as designed. <p>What is the (1) status of alarm D4931, "ECCS & CBS Recirc Initiated" and (2) why?</p> <table style="width: 100%; border: none;"> <tr> <td style="width: 50%; text-align: center;">(1)</td> <td style="width: 50%; text-align: center;">(2)</td> </tr> <tr> <td>A. In Alarm</td> <td>RWST level is below the recirc swap over setpoint.</td> </tr> <tr> <td>B. In Alarm</td> <td>Containment sump level is above the recirc swap over setpoint.</td> </tr> <tr> <td>C. Reset</td> <td>RWST level is above the recirc swap over setpoint.</td> </tr> <tr> <td>D. Reset</td> <td>SI reset will require manual operator action to initiate cold leg recirculation.</td> </tr> </table>				(1)	(2)	A. In Alarm	RWST level is below the recirc swap over setpoint.	B. In Alarm	Containment sump level is above the recirc swap over setpoint.	C. Reset	RWST level is above the recirc swap over setpoint.	D. Reset	SI reset will require manual operator action to initiate cold leg recirculation.
(1)	(2)												
A. In Alarm	RWST level is below the recirc swap over setpoint.												
B. In Alarm	Containment sump level is above the recirc swap over setpoint.												
C. Reset	RWST level is above the recirc swap over setpoint.												
D. Reset	SI reset will require manual operator action to initiate cold leg recirculation.												
Proposed Answer:	A.												
Explanation (Optional):													
<p>A is correct. Conditions for D4931 to be in alarm are: RWST level <120478 gallons and SI signal present. Therefore, the alarm is in due to RWST level being below the setpoint. The SI has been reset as expected to allow for equipment realignment. With no equipment malfunctions, the SI signal will still exist for cold leg recirculation mode. The SI signal has a separate reset switch for cold leg recirculation mode. This switch is labeled "S SIGNAL RESET</p>													

<p>FOR S/RWST LO-LO CBS-V8 or CBS-V14 AUTO OPEN". This switch is only operated for a loss of recirculation capability.</p> <p>B is incorrect but plausible. Conditions for D4931 to be in alarm are: RWST level <120478 gallons and SI signal present. Therefore, the alarm is in due to RWST level being below the setpoint. Injecting RWST contents will cause an increase in recirc sump level however, this is not the initiator of the semi-automatic swap over.</p> <p>C is incorrect but plausible. If the student is not able to verify the status of the alarm because the setpoint is unknown, this is a possible answer.</p> <p>D is incorrect but plausible. The SI has been reset as expected to allow for equipment realignment. With no equipment malfunctions, the SI signal will still exist for cold leg recirculation mode. The SI signal has a separate reset switch for cold leg recirculation mode. This switch is labeled "S SIGNAL RESET FOR S/RWST LO-LO CBS-V8 or CBS-V14 AUTO OPEN". This switch is only operated for a loss of recirculation capability.</p>			
<p>Technical Reference(s):</p>		<p>VPRO for D4931</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1203I 06</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		
	<p>Comprehension or Analysis</p>		<p>x</p>
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>	<p>(10)</p>	
	<p>55.43</p>		
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q76	Tier #		1
	Group #		1
	K/A #	000007 (EPE 7; BW E02&E10; CE E02) Reactor Trip, Stabilization, Recovery / 1 EA2.02 Ability to determine or interpret the following as they apply to a reactor trip: Proper actions to be taken if the automatic safety functions have not taken place.	
	Importance Rating		4.6
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The crew has implemented FR-S.1, "Response to Nuclear Power Generation/ATWS" and is at Step 15, 'Verify Reactor Subcritical'. • Control rods will not insert in Auto or Manual. • Boration flow cannot be established to the Reactor Coolant System. • Power Range NI channels are fluctuating between 10-15% power. • Tavg is 600°F and slowly increasing. • All Steam Generator Narrow Range Levels are 10% and stable. • Total EFW flow is throttled to 400 gpm. <p>What procedural actions are required in response to these conditions?</p> <p>A. Allow the RCS to heat up and transition to E-0, 'Reactor Trip or Safety Injection'.</p> <p>B. Remain in FR-S.1 and maximize feed flow to cool down and depressurize the RCS until boration flow is established.</p> <p>C. Transition to FR-C.1, 'Response to Inadequate Core Cooling' to minimize cooldown of the RCS. Return to FR-S.1 when boration flow is established.</p> <p>D. Allow the RCS to heat up. Perform actions of other Functional Restoration Procedures in effect which do not cooldown the RCS. Return to Step 4 of FR-S.1.</p>			

Proposed Answer:		D.
Explanation (Optional):		
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5) Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate is required to assess plant conditions, determine what procedure transitions are required and allowed and what actions required to add negative reactivity.</p> <p>D is correct. FR-S.1, Step 15 RNO states “Continue to borate. <u>IF</u> boration is <u>NOT</u> available, <u>THEN</u> allow the RCS to heat up. Perform actions of other Functional Restoration Procedures in effect which do not cooldown or otherwise add positive reactivity to the core. Return to Step 4.”</p> <p>A is incorrect but plausible. FR-S.1 does direct allowing the RCS to heat up for these conditions, however, a transition to E-0 at this point is not correct. Step 15 RNO directs implementing and applicable FRP’s and returning to Step 4 of FR-S.1. A return to Step 4 of FR-S.1 facilitates re-evaluation of reactor trip conditions, at which time a transition to procedure and step in effect would be appropriate.</p> <p>B is incorrect but plausible. It is correct that FR-S.1 would still be in effect, however the reactor is not subcritical at this point and it is desirable to allow the RCS to heat up in order to introduce negative reactivity. Core cooling is a major concern for an ATWS event, so it is conceivable that actions could be taken to address core cooling concerns, however the introduction of negative reactivity is of higher priority. Depressurizing the RCS is a strategy earlier in FR-S.1 if boration flow is inadequate. In this case the question stem indicates that boration flow cannot be established for an unspecified reason.</p> <p>C is incorrect but plausible. Core cooling is a major concern for an ATWS event, so it is conceivable that actions could be taken to address core cooling concerns, however the introduction of negative reactivity is of higher priority. FR-S.1, Step 15 RNO does discuss transitioning to other FRP procedures, however it states that they should not cooldown or otherwise add positive reactivity to the core. Returning to step 4 of FR-S.1 is directed by the RNO, not when boration is established.</p>		
Technical Reference(s):	FR-S.1, “Response to Nuclear Power Generation/ATWS” Rev 30	
Proposed references to be provided to applicants during examination:	None	
Learning Objective:	SBK LOP L1200I 02	

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 ES-401-5 Written Examination Question Worksheet

Question Source:	Bank #	x	TEB 32483	
	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:		2009 Seabrook NRC Remediation Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41			
	55.43	(5), (6)		
Comments:				

A is correct. Per E-1, "Loss of Reactor or Secondary Coolant" swap to hot leg recirc is made after 5 hours. Hot leg recirculation swap is performed per ES-1.4. Hot leg recirculation is performed in order to prevent boron precipitation from impeding heat transfer from the fuel to the coolant.

B is incorrect but plausible. After 4 hours, preparations are made for the transition to hot leg recirc, although the actual transition is not made until 5 hours after the event.

C is incorrect but plausible. ES-1.2 is a recovery procedure for a LOCA in which RCS pressure remains above the shutoff head of the RHR pumps. Because of the name of the procedure it is a plausible distracter. Part (1) is correct.

D is incorrect but plausible. After 4 hours, preparations are made for the transition to hot leg recirc, although the actual transition is not made until 5 hours after the event. ES-1.2 is a recovery procedure for a LOCA in which RCS pressure remains above the shutoff head of the RHR pumps. Because of the name of the procedure it is a plausible distracter.

Technical Reference(s):	E-1, "Loss of Reactor or Secondary Coolant", Rev 44		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1203I 09		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO															
Q78	Tier #		1															
	Group #		1															
	K/A #	000025 (APE 25) Loss of Residual Heat Removal System / 4 AA2.05 Ability to determine and interpret the following as they apply to the Loss of Residual Heat Removal System: Limitations on LPI flow and temperature rates of change.																
	Importance Rating		3.5															
Proposed Question:																		
<p>Plant conditions:</p> <p>The crew is responding to a loss of RHR using OS1213.02, "Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions" step 5.</p> <ul style="list-style-type: none"> • RCS level is (-)82". • The reactor has been shut down for 5 days. • RCS temperature was 100 °F when RHR cooling was lost. • PRA has not provided a time to boil value. <p>What is the time to boiling (1) and once an RHR pump is started, what will be the maximum allowed flow if RCS level is not changed (2)?</p> <p>(reference provided)</p> <table style="width: 100%; border: none;"> <thead> <tr> <th></th> <th style="text-align: center;">(1)</th> <th style="text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A.</td> <td>18 minutes</td> <td>3000 gpm</td> </tr> <tr> <td>B.</td> <td>14 minutes</td> <td>3000 gpm</td> </tr> <tr> <td>C.</td> <td>18 minutes</td> <td>3500 gpm</td> </tr> <tr> <td>D.</td> <td>14 minutes</td> <td>3500 gpm</td> </tr> </tbody> </table>					(1)	(2)	A.	18 minutes	3000 gpm	B.	14 minutes	3000 gpm	C.	18 minutes	3500 gpm	D.	14 minutes	3500 gpm
	(1)	(2)																
A.	18 minutes	3000 gpm																
B.	14 minutes	3000 gpm																
C.	18 minutes	3500 gpm																
D.	14 minutes	3500 gpm																

Proposed Answer:	B.	
Explanation (Optional):		
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must correctly select and implement figures in the given procedure. These figures are effectively used as attachments.</p> <p>B is correct. A, C and D are incorrect but plausible. To correctly obtain (1) the student must select OS1213.02-4 and read up from 120 hours (5 days) to the given 100 °F curve to obtain 14 minutes. If the student incorrectly used OS1213.02-5, 18 minutes would be chosen making this plausible. To correctly obtain (2) the student must read from -82" to the pump cavitation line. The maximum flow for this level is 3000 gpm. If the student selects the minimum level for 3500 gpm as labeled, this is a plausible but incorrect answer, particularly because normal RHR flow is 3500 gpm.</p>		
Technical Reference(s):	OS1213.01, "Loss of RHR While Operating at Reduced Inventory or Mid-Loop Conditions" Rev 18	
Proposed references to be provided to applicants during examination:	OS1213.02 pages 5, 40-42	
Learning Objective:	SBK LOP L1705I 02, 03	
Question Source:	Bank #	
	Modified Bank#	(Note changes or attach Parent)
	New	x
Question History:		
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	x
10 CFR Part 55 Content:	55.41	
	55.43	(5)
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q79	Tier #		1
	Group #		1
	K/A #	000026 (APE 26) Loss of Component Cooling Water / 8 2.4.47 Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material.	
	Importance Rating		4.2

Proposed Question:

Plant conditions:

- The crew notices that the 'A' PCCW Head Tank Level is decreasing.
- Hardwire alarm on UA-50 "PCCW Head Tank 'A' Level LO" is in alarm.
- The crew has entered procedure OS1212.01, 'PCCW System Malfunction'.
- 'A' PCCW Head Tank Level is as shown on the MPCs trend below:



What actions will the US direct in accordance with OS1212.01?

(reference provided)		
<p>A. Trip the reactor, enter E-0. Trip the RCPs.</p> <p>B. Trip the reactor, enter E-0. Isolate PCCW to the Waste Process Building, Spent Fuel Pool Heat Exchanger, and Rad Monitor.</p> <p>C. Locally make up to the head tank. Locate and isolate the leak if possible. Check the 'A' PCCW heat exchanger outlet temperatures 65°F to 75°F.</p> <p>D. Locally make up to the head tank. Locate and isolate the leak if possible. Isolate PCCW to the Waste Process Building, Spent Fuel Pool Heat Exchanger, and Rad Monitor.</p>		
Proposed Answer:	D.	
Explanation (Optional):		
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions based partly upon hardwire alarms and computer system indications and select the correct procedural strategy.</p> <p>D is correct. PCCW is designed to auto isolate to the WPB and PCCW rad monitor @ 42% level. Procedural step 5f directs isolating PCCW to the Waste Process Building, Spent Fuel Pool Heat Exchanger, and RDMS.</p> <p>A is incorrect but plausible. OS1212.01, step 5g directs tripping the reactor and securing the affected RCP's if head tank level drops below 36% level.</p> <p>B is incorrect but plausible. OS1212.01, step 5g directs tripping the reactor and securing the affected RCP's if head tank level drops below 36% level. Loads given will be isolated, but not after the reactor is tripped.</p> <p>C is incorrect but plausible. OS1212.01, step 5 does direct locally making up to the head tank, locating the leak, and isolating if possible. Direction for checking PCCW heat exchanger outlet temperature in the normal band is included in OS1212.01, however it is associated with actions in response to degraded PCCW cooling conditions.</p>		
Technical Reference(s):	OS1212.01, "PCCW System Malfunction" Rev 14	
Proposed references to be provided to applicants during examination:	None	
Learning Objective:	SBK LOP L1445I 02	

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Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	x		
Question History:				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41			
	55.43	(5)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q80	Tier #		1
	Group #		1
	K/A #	000056 (APE 56) Loss of Offsite Power / 6 2.4.44 Knowledge of emergency plan protective action recommendations.	
	Importance Rating		4.6
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The plant was at 100% power. • A loss of all offsite AC power has occurred. • Both Emergency Diesels are damaged and cannot be started. • SEPS is unavailable. • Core Exit Thermocouple temperatures are 1150 °F and rising. • Post LOCA rad monitors RM-6576A-1 and 6576B-1 are reading 20 R/hr and rising. • A General Emergency has been declared on MG1. • No radioactive release has occurred. • There was no previous GE PAR issued. <p>What is the required PAR determination? (reference provided)</p> <p>A. PAR 'A' B. PAR 'B' C. PAR 'C' D. Run Raddose V to determine PAR 'A' or 'B'</p>			
Proposed Answer:	B.		

Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess the given plant conditions and determine the protective action recommendations that apply.</p> <p>B is correct. Per figure 2 of ER-5.4, PAR 'B' applies. A GE was declared, because POST LOCA Rad monitors are < 1305 R/hr and that the GE was declared on MG1 and no release is yet in progress (i.e. RG1 is not declared), the first decision block on ER 5.4 is "no". Hostile action is not in progress. A red path for core cooling exists based upon CETC >1100°F. This leads to PAR 'B'.</p> <p>A and C are incorrect but plausible. If the student cannot correctly follow the decision tree with the given information PAR 'A' or 'C' may be reached.</p> <p>D is incorrect but plausible. A Raddose V run must be performed only if a release is in progress to determine if PAR 'A' or 'B' applies.</p>			
Technical Reference(s):		ER 5.4 Rev 38 F-0.2 Core Cooling (C) Rev 20	
Proposed references to be provided to applicants during examination:			ER 5.4 Figure 2 F-0.2
Learning Objective:	SBK LOP L1308I 01		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		

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	55.43	(5)
Comments:		

Examination Outline Cross-reference:		Level	RO	SRO															
Q81		Tier #		1															
		Group #		1															
		K/A #	(W E04) LOCA Outside Containment / 3 2.4.18 Knowledge of the specific bases for EOPs.																
		Importance Rating		4.0															
Proposed Question:																			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Reactor trip and safety injection. • While processing E-0, "Reactor Trip or Safety Injection" step 20, "Check for Leakage Outside Containment", the BOP reports that Radiation monitor "HI RANGE RHR VAULT TR A" is in alarm. • After dispatching NSOs and RP, RCS leakage is identified in the 'A' RHR vault. <p>What procedure transition is first required (1) and in accordance with that procedure, why (2)?</p> <p>ECA-1.1, "Loss of Emergency Coolant Recirculation" ECA-1.2, "LOCA Outside Containment"</p> <table style="width: 100%; border: none;"> <thead> <tr> <th style="width: 10%;"></th> <th style="width: 20%; text-align: center;">(1)</th> <th style="width: 70%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A.</td> <td>ECA-1.2</td> <td>Procedure provides actions to identify and isolate a LOCA outside containment.</td> </tr> <tr> <td>B.</td> <td>ECA-1.1</td> <td>Procedure provides actions to identify and isolate a LOCA outside containment.</td> </tr> <tr> <td>C.</td> <td>ECA-1.2</td> <td>A loss of RCS inventory to the RHR vaults will compromise emergency coolant recirculation capability.</td> </tr> <tr> <td>D.</td> <td>ECA-1.1</td> <td>A loss of RCS inventory to the RHR vaults will compromise emergency coolant recirculation capability.</td> </tr> </tbody> </table>						(1)	(2)	A.	ECA-1.2	Procedure provides actions to identify and isolate a LOCA outside containment.	B.	ECA-1.1	Procedure provides actions to identify and isolate a LOCA outside containment.	C.	ECA-1.2	A loss of RCS inventory to the RHR vaults will compromise emergency coolant recirculation capability.	D.	ECA-1.1	A loss of RCS inventory to the RHR vaults will compromise emergency coolant recirculation capability.
	(1)	(2)																	
A.	ECA-1.2	Procedure provides actions to identify and isolate a LOCA outside containment.																	
B.	ECA-1.1	Procedure provides actions to identify and isolate a LOCA outside containment.																	
C.	ECA-1.2	A loss of RCS inventory to the RHR vaults will compromise emergency coolant recirculation capability.																	
D.	ECA-1.1	A loss of RCS inventory to the RHR vaults will compromise emergency coolant recirculation capability.																	
Proposed Answer:	A.																		

Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem and apply detailed knowledge of procedure transitions from E-0 to select the correct procedure and its basis.</p> <p>A is correct. Step 20 of E-0 will evaluate conditions outside of containment. With high radiation conditions in 'A' RHR vault, E-0 will transition to ECA-1.2 to attempt to isolate the LOCA outside of containment.</p> <p>B is incorrect but plausible. ECA-1.2 will transition to ECA-1.1 if the leakage outside of containment cannot be isolated.</p> <p>C is incorrect but plausible. ECA-1.2 is the correct transition, however the basis given is for ECA-1.1. If the LOCA outside containment cannot be isolated the concern is compromising ECCS recirculation capability.</p> <p>D is incorrect but plausible. ECA-1.2 will transition to ECA-1.1 if the leakage outside of containment cannot be isolated. If the LOCA outside containment cannot be isolated the concern is compromising ECCS recirculation capability.</p>			
Technical Reference(s):		E-0, "Reactor Trip or Safety Injection" Rev 57	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1209I 04, 06		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41		

2020 Seabrook Station NRC Written Exam
ES-401-5 Written Examination Question Worksheet

	55.43	(5)
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q82	Tier #		1
	Group #		2
	K/A #	000003 (APE 3) Dropped Control Rod / 1 AA2.03 Ability to determine and interpret the following as they apply to the Dropped Control Rod: Dropped rod, using in-core/ex-core instrumentation, in-core or loop temperature measurements.	
	Importance Rating		3.8
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Crew is implementing OS1000.07, "Approach to Criticality" • Plant startup with control rods is in progress. • The reactor is critical with a positive startup rate of 0.4 DPM. • Intermediate range power is 10^{-9} amps. <p>The following indications are received:</p> <ul style="list-style-type: none"> • Rod H8 rod bottom light is lit. • DRPI rod deviation lights are lit. • Startup rate is now (-)0.3 DPM. 			

What is the required abnormal procedure **(1)** and what action will the Unit Supervisor take **(2)**?

- | (1) | (2) |
|----------------------------------|--|
| A. OS1210.05, "Dropped Rod". | Direct the PSO to stabilize reactor power. |
| B. OS1210.05, "Dropped Rod". | Use OS1000.03, "Plant Shutdown from Minimum Load to Hot Standby" to place the plant in Mode 3. |
| C. OS1210.07, "RPI Malfunction". | Direct the PSO to stabilize reactor power. |
| D. OS1210.07, "RPI Malfunction". | Use OS1000.03, "Plant Shutdown from Minimum Load to Hot Standby" to place the plant in Mode 3. |

Proposed Answer:

B .

Explanation (Optional):

SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem to determine that a control rod has dropped vs a DRPI malfunction. Once the correct procedure is selected, the candidate must rely on knowledge of the content of this procedure and the based on the current plant conditions determine that a plant shutdown is required.

B is correct. The indications as given are for a dropped control rod. OS1210.05 will evaluate is the reactor is critical. If the reactor is critical the rod will be recovered immediately, if not critical the plant will be placed in Mode 3 using OS1000.03 Plant Shutdown from Minimum Load to Hot Standby" and the rod will be recovered during a subsequent startup.

A is incorrect but plausible. The DRPI indications as given are for a dropped control rod. OS1210.05 will evaluate is the reactor is critical. If the reactor is critical the rod will be recovered immediately, if not critical the plant will be placed in Mode 3 using OS1000.03 Plant Shutdown from Minimum Load to Hot Standby" and the rod will be recovered during a subsequent startup. It is plausible that because the plant is in mode 2 the only requirement is to stabilize power.

C is incorrect but plausible. The indications of dropped control rods and DRPI malfunctions are a common source of misconception. It is plausible that the indications given are for a DRPI malfunction. OS1210.07 will have the crew stop power change evolutions per step 1b RNO if a DRPI malfunction is in progress.

D is incorrect but plausible. The indications of dropped control rods and DRPI malfunctions are a common source of misconception. It is plausible that the indications given are for a DRPI malfunction. OS1210.07 power to be stabilized but it is plausible that the procedure requires placing the plant in mode 3 as the DRPI requirements are less restrictive in mode 3 vs mode 2.

2020 Seabrook Station NRC Written Exam
 ES-401-5 Written Examination Question Worksheet

Technical Reference(s):		OS1210.05, "Dropped Rod", Rev 16	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1185I 04		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q83	Tier #		1
	Group #		2
	K/A #	000028 (APE 28) Pressurizer (PZR) Level Control Malfunction / 2 AA2.01 Ability to determine and interpret the following as they apply to the Pressurizer Level Control Malfunctions: PZR level indicators and alarms.	
	Importance Rating		3.6
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • Pressurizer level channels 459/460 are selected for control and backup. • A failure occurs on pressurizer level channel 461. <p>The PSO reports the following indications:</p> <ul style="list-style-type: none"> • RC-LI-459 = 60% stable • RC-LI-460 = 60% stable • RC-LI-461 = 100% stable. <p>The US has begun implementing OS1201.07, "PZR Level Instrument Failure".</p> <p>What actions are required?</p> <p>A. Direct the PSO to increase charging flow to restore pressurizer level per OS1201.07, "PZR Level Instrument Failure".</p> <p>B. Place the failed channel in the tripped condition within 6 hours per TS 3.3.1 action 6.</p> <p>C. Direct the PSO to restore letdown per OS1201.07, "PZR Level Instrument Failure".</p> <p>D. Place the failed channel in bypass within 6 hours per TS 3.3.1 action 6.</p>			

Proposed Answer:		B.		
Explanation (Optional):				
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(2) facility operating limitations in the Technical Specifications and their bases. The Candidate is required to evaluate the malfunction that has occurred and correctly determine the required TS action.</p> <p>B is correct. With channels 459/460 selected for control and backup, channel 461 acts a protection channel only. Based upon the indications reported, 461 failed high. Procedure OS1201.07, "PZR Level Instrument Failure" will be entered. With the failed channel, TS 3.3.1 will require the associated bistables to be placed in the tripped conditions.</p> <p>A is incorrect but plausible. If the channel were a control channel, it failing high would have caused a reduction in charging flow. With the given channel alignment however, no automatic action will occur.</p> <p>C is incorrect but plausible. If the channel were a control channel, it failing low would have caused a letdown isolation. With the given channel alignment however and the fact that it failed high, no automatic action will occur.</p> <p>D is incorrect but plausible. Failed channels may be placed in bypass for required testing of redundant channels however, this is not the required TS action.</p>				
Technical Reference(s):		OS1201.07, "PZR Level Instrument Failure" Rev 17		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L1182 03			
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	x		
Question History:				
Question Cognitive	Memory or Fundamental Knowledge			

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Level:	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41			
	55.43	(2)		
Comments:				

Examination Outline Cross-reference:		Level	RO	SRO
Q84		Tier #		1
		Group #		2
		K/A #	000068 (APE 68; BW A06) Control Room Evacuation / 8 2.4.27 Knowledge of "fire in the plant" procedures.	
		Importance Rating		3.9
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 100% power. • A fire has been confirmed in the cable spreading room. • The crew is responding using OS1200.00, "Response to Fire or Fire Alarm Actuation". • Prompt Actions for the Affected Fire Area have been performed per Attachment 'C'. • Control room evacuation will be accomplished using OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown Facilities". <p>How will adequate heat removal be provided in OS1200.02 as the crew travels to the RSS panels?</p> <p>A. EFW flow will be maintained greater than 500 gpm total to all SGs. B. The condenser steam dump valves will modulate open in the Tavg mode. C. The condenser steam dump valves will modulate open in the steam pressure mode. D. Steam header pressure will be allowed to increase to the steam generator safety valve setpoints.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. This question relies on the candidate's ability to assess the given plant conditions and understand how the procedure strategy and attachments will impact plant response.</p>				

<p>D is correct. The MSIVs have been closed procedurally. The ASDVs are closed by performing Prompt Actions per attachment 'C' and will no longer automatically open after paced in close. The SRO candidate must understand and apply this knowledge of the attachment. The RCS temperature will rise until the steam header pressure increases to the steam generator safety valve setpoint.</p> <p>A is incorrect but plausible. Sufficient inventory is available in the steam generators until the remote safe shutdown panels are manned. The EFW system is not checked until the panels are manned.</p> <p>B is incorrect but plausible. The condenser steam dumps are isolated due to the closure of the MSIVs.</p> <p>C is incorrect but plausible. The condenser steam dumps are isolated due to closure of the MSIVs.</p>			
Technical Reference(s):		OS1200.00, "Response to Fire or Fire Alarm Actuation" Rev 25 OS1200.02, "Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities" Rev 23	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK L8210I 02		
Question Source:	Bank #	<input checked="" type="checkbox"/>	10728
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2009 Seabrook NRC Exam SRO Section		
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		<input checked="" type="checkbox"/>
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q85		Tier #		1
		Group #		2
		K/A #	(BW E09; CE A13**; W E09 & E10) Natural Circulation/4 2.2.44 Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.	
		Importance Rating		4.4
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A natural circulation cooldown is in progress per ES-0.2, "Natural Circulation Cooldown". • The crew is preparing to initiate RCS depressurization. • No CRDM fans are running. • There are no inactive RCS loops. <p>What restrictions apply to the cooldown and depressurization?</p> <p>A. The cooldown rate is limited to <30 °F/hr. No depressurization is allowed without at least 1 CRDM fan running.</p> <p>B. Maintain subcooling 100 – 130 °F. No depressurization is allowed without at least 1 CRDM fan running.</p> <p>C. The cooldown rate is limited to <30 °F/hr. Depressurization is permitted after 88 hours.</p> <p>D. Maintain subcooling 100 – 130 °F. Depressurization is permitted after 88 hours.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and				

emergency situations. This question relies on the candidate's ability to assess the given plant conditions and select the required procedural actions in terms of RCS temperature and pressure relationship and of the ability to depressurize or not.

D is correct. With no CRDM fans running ES-0.2 will impose additional subcooling requirements of 100-130 °F. Depressurization is permitted but only after the reactor vessel head is allowed to cool for 88 hours.

A is incorrect but plausible. The expected cooldown rate in ES-0.2 is 30-50 °F/hr with two CRDM fans running. It is plausible that having no CRDM fans running would limit this to <30 °F/hr. It is conceivable that no depressurization would be permitted due to the concerns of reactor vessel head voiding.

B is incorrect but plausible. With no CRDM fans running ES-0.2 will impose additional subcooling requirements of 100-30 °F. It is conceivable that no depressurization would be permitted due to the concerns of reactor vessel head voiding.

C is incorrect but plausible. The expected cooldown rate in ES-0.2 is 30-50 °F/hr with two CRDM fans running. It is plausible that having no CRDM fans running would limit this to <30 °F/hr. Depressurization is permitted but only after the reactor vessel head is allowed to cool for 88 hours.

Technical Reference(s):	ES-0.2, "Natural Circulation Cooldown" Rev 38		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1225I 06		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41		

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	55.43	(5)
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q86	Tier #		2
	Group #		1
	K/A #	003 (SF4P RCP) Reactor Coolant Pump A2.03 Ability to (a) predict the impacts of the following malfunctions or operations on the RCPS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Problems associated with RCP motors, including faulty motors and current, and winding and bearing temperature problems.	
	Importance Rating		3.1
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Plant power is at 30% for a chemistry hold. • A malfunction in the 'D' RCP has occurred. • Alarm B7091, "RCP D MTR STATOR WINDING TEMP HI-HI" has occurred. • A0730, "RCP D MTR STAT WDG TEMP" is reading 305 °F and rising. <p>What abnormal procedure is required (1) and what actions will the crew take prior to stopping the 'D' RCP (2)?</p>			

(1)	(2)
A. OS1212.01, "PCCW System Malfunction"	Commence feeding the 'D' SG, defeat loop temperature inputs.
B. OS1212.01, "PCCW System Malfunction"	Trip the reactor, go to E-0.
C. OS1201.01, "RCP Malfunction".	Commence feeding the 'D' SG, defeat loop temperature inputs.
D. OS1201.01, "RCP Malfunction".	Trip the reactor, go to E-0.
Proposed Answer:	C.
Explanation (Optional):	
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem to select the correct AOP and determine what actions are required to mitigate the consequences of the equipment failure.</p> <p>C is correct. The conditions as given meet the entry criteria for OS1201.01. With power below P-8 (50%) the reactor will not be tripped before the 'D' RCP is stopped.</p> <p>A is incorrect but plausible. The RCP motors are cooled by PCCW. It is plausible that the OS1212.01 would contain actions to mitigate the effects of the high stator winding temperatures however, the entry conditions are not met. OS1201.01 is required.</p> <p>B is incorrect but plausible. The RCP motors are cooled by PCCW. It is plausible that the OS1212.01 would contain actions to mitigate the effects of the high stator winding temperatures however, the entry conditions are not met. OS1201.01 is required.</p> <p>D is incorrect but plausible. The conditions as given meet the entry criteria for OS1201.01. With power below P-8 (50%) the reactor will not be tripped before the 'D' RCP is stopped.</p>	
Technical Reference(s):	OS1201.01, "RCP Malfunction" Rev 19
Proposed references to be provided to applicants during examination:	None
Learning Objective:	SBK LOP L1181I 02, 03

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Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	x		
Question History:				
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41			
	55.43	(5)		
Comments: Alarm setpoint for B7091 is 302 °F.				

Examination Outline Cross-reference:	Level	RO	SRO
Q87	Tier #		2
	Group #		1
	K/A #	007 (SF5 PRTS) Pressurizer Relief/Quench Tank 2.4.2 Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.	
	Importance Rating		4.6
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The crew has initiated a safety injection in response to a steam generator tube rupture. • A loss of all offsite power occurred following the safety injection. • The crew is preparing to perform the initial depressurization of the RCS using a PORV. • A caution in E-3, "Steam Generator Tube Rupture" states: <ul style="list-style-type: none"> ○ "The PRT may rupture if a PZR PORV is used to depressurize the RCS. This may result in abnormal containment conditions". <p>In accordance with the E-3 background document, which of the following statements describes the implications of this caution?</p> <p>A. The PRT rupture disk may fail before RCS pressure is reduced to ruptured SG pressure. This will result in increasing containment radiation and humidity. The crew should transition to E-1 if this occurs.</p> <p>B. The PRT rupture disk may fail before RCS pressure is reduced to ruptured SG pressure. This will result in increasing containment radiation and humidity. The crew should continue recovery in this guideline unless otherwise directed in E-3.</p> <p>C. Cycling of the PZR PORV should be minimized to avoid failure of the PRT rupture disc. Do not use the PORV if PRT rupture disc failure is imminent. The crew should transition to ECA-3.3, "SGTR Without Pressurizer Pressure Control".</p> <p>D. Cycling of the PZR PORV should be minimized to avoid failure of the PRT rupture disc. Use of auxiliary spray is preferred over use of a PORV. The crew should transition to ECA-3.3, "SGTR</p>			

Without Pressurizer Pressure Control”.			
Proposed Answer:	B.		
Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must utilize detailed system knowledge of the PRT and select the correct procedural strategy.</p> <p>B is correct. While the PRT rupture disc may fail before RCS pressure is reduced to ruptured SG pressure, the crew should utilize this method as directed. The caution is alerting them that the abnormal containment conditions could be from this source, as opposed to a separate RCS leak. The crew should continue recovery in E-3 unless the conditions degrade such that a transition to another procedure is required.</p> <p>A is incorrect but plausible. It is true that the PRT rupture disc may fail before RCS pressure is reduced to rupture SG pressure and that this may result in increasing containment radiation and humidity. The crew would only transition if RCS subcooling or pressurizer level cannot be maintained and the transition would be to ECA-3.1.</p> <p>C is incorrect but plausible. The caution prior to step 18 does warn that cycling of the PORV should be minimized to avoid failure of the rupture disc, but the background document states that this is to minimize the chance of failure of a PORV. A transition to ECA-3.3 would be required if the crew determined that a PORV and auxiliary spray was not available because the rupture disc was going to fail.</p> <p>D is incorrect but plausible. The RNO for step 18 would only use auxiliary spray of a PORV was not available. No consideration is made for preventing a rupture of the PRT. The crew would only transition to ECA-3.3 if no PORV or auxiliary spray were available.</p>			
Technical Reference(s):		E-3, “Steam Generator Tube Rupture” Rev 45 Background document for E-3, Rev 3.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8022I 11 SBK LOP 1205I 03		
Question Source:	Bank #	x	9476

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	Modified Bank#			(Note changes or attach Parent)
	New			
Question History:		2009 Seabrook NRC Exam		
Question Cognitive Level:	Memory or Fundamental Knowledge			
	Comprehension or Analysis		x	
10 CFR Part 55 Content:	55.41			
	55.43	(5)		
Comments:				

Examination Outline Cross-reference:	Level	RO	SRO
Q88	Tier #		2
	Group #		1
	K/A #	010 (SF3 PZR PCS) Pressurizer Pressure Control A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures	
	Importance Rating		3.6
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • 30% power. • The 'D' group of backup heaters are tagged out for maintenance and unavailable. • Pressurizer sprays are being forced with the 'A', 'B' and 'C' groups of backup heaters energized. • The 'C' backup heaters trip due a breaker malfunction. • RC-PCV-455A "C' RCP Spray Valve" fails as-is and does not close. • Pressurizer pressure is 2,000 psig and lowering. <p>Per OS1201.06, "PZR Pressure Instrument/Component Failure" what actions will the US direct?</p> <p>A. Stop the 'C' RCP per OS1201.06 <u>only</u>.</p> <p>B. Stop the 'C' RCP per OS1201.06, then trip the reactor and go to E-0.</p> <p>C. Trip the reactor and go to E-0, when directed by ES-0.1 implement OS1201.06 to stop the 'C' RCP.</p> <p>D. Trip the reactor and go to E-0, when immediate actions are complete, then stop the 'C' RCP per OS1201.06.</p>			

Proposed Answer:		D.	
Explanation (Optional):			
<p>D is correct. Per step 2 RNO of OS1201.06, "PZR Pressure Instrument/Component Failure" if RCS pressure continues to decrease with a failed spray valve, the reactor will be tripped, transition to E-0, and once immediate actions are complete, the RCP supplying the failed spray valve will be stopped per OS1201.06. The AOP and EOP must be coordinated in parallel.</p> <p>A is incorrect but plausible. With power less than 50% (P-9) it is plausible that only stopping the RCP supplying the failed spray valve would be required to mitigate the pressure decrease. However, there is no consideration of power level in OS1201.06. The reactor will be tripped regardless of pressure.</p> <p>B is incorrect but plausible. With power less than 50% (P-9) it is plausible that stopping the RCP supplying the failed spray valve would be performed before tripping the reactor. However, there is no consideration of power level in OS1201.06. The reactor will be tripped regardless of pressure.</p> <p>C is incorrect but plausible. ES-0.1 step 7 directs the crew to "evaluate implementation of abnormal operating procedures". This step is intended to remind the crew that now is an appropriate time to implement AOPs that would otherwise interrupt the strategy being implemented by the EOP. This is contrary to the strategy of OS1201.06.</p>			
SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must demonstrate the knowledge of the correct actions to take and how to coordinate parallel implementation of EOPs and AOPs.			
Technical Reference(s):		OS1201.06, "PZR Pressure Instrument/Component Failure" Rev 15	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1182I 05		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			

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

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Examination Outline Cross-reference:	Level	RO	SRO
Q89	Tier #		2
	Group #		1
	K/A #	026 (SF5 CSS) Containment Spray 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:

- LOCA inside containment.
- ECA-1.1, "Loss of Emergency Coolant Recirculation" is in progress due to the loss of both trains of RHR.
- A valid ORANGE condition arises on the containment (Z) critical safety function status tree.
- RWST level is 250,000 gallons and decreasing.
- Containment pressure is 20 psig and rising.
- All containment Phase 'A' and 'B' penetrations are isolated.

What action is required **(1)** and what operational limitations apply to the CBS pumps **(2)**?

(1)	(2)
A. Transition to FR-Z.1, "Response to High Containment Pressure".	One CBS pump should be running as directed by ECA-1.1.
B. Transition to FR-Z.1, "Response to High Containment Pressure".	Both CBS pumps should be left running until containment pressure decreases to less than 18 psig.
C. Remain in ECA-1.1, "Loss of Emergency Coolant Recirculation".	One CBS pump should be running as directed by ECA-1.1.
D. Remain in ECA-1.1, "Loss of Emergency Coolant Recirculation".	Both CBS pumps should be left running until containment pressure decreases to less than 18 psig.

Proposed Answer:	A.	
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Explanation (Optional):

SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem to select the correct procedure and understand the organizational relationship between the two possibilities regarding operation of the CBS pumps.

A is correct. Based upon the conditions in the stem, while processing ECA-1.1 the orange condition on Z requires transition to FR-Z.1. Guidance on how to operate CBS pumps is contained in a note in FR-Z.1, "If ECA 1.1, LOSS OF EMERGENCY COOLANT RECIRCULATION, is in effect, containment spray should be operated as directed in ECA 1.1 rather than step 2 below."

<p>B is incorrect but plausible. Based upon the conditions in the stem, while processing ECA-1.1 the orange condition on Z requires transition to FR-Z.1. The CBS pumps should be run in accordance with ECA-1.1 not FR-Z.1. With containment pressure between 18 and 52 psig, one CBS pump will be left running.</p> <p>C is incorrect but plausible. It is plausible that performance of ECA-1.1 would require completion before transitioning as is required in ES-1.3. This is particularly true because of the few actions that will be taken in FR-Z.1.</p> <p>D is incorrect but plausible. It is plausible that performance of ECA-1.1 would require completion before transitioning as is required in ES-1.3. This is particularly true because of the few actions that will be taken in FR-Z.1. The CBS pumps should be run in accordance with ECA-1.1 not FR-Z.1. With containment pressure between 18 and 52 psig, one CBS pump will be left running.</p>			
Technical Reference(s):		ECA-1.1, "Loss of Emergency Coolant Recirculation" Rev 38. FR-Z.1, "Response to High Containment Pressure" Rev 23.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1212I 08		
Question Source:	Bank #	<input type="checkbox"/>	<input type="checkbox"/>
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input checked="" type="checkbox"/>	<input type="checkbox"/>
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		<input type="checkbox"/>
	Comprehension or Analysis		<input checked="" type="checkbox"/>
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

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Examination Outline Cross-reference:	Level	RO	SRO
Q90	Tier #		2
	Group #		1
	K/A #	073 (SF7 PRM) Process Radiation Monitoring 2.4.21 Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.	
	Importance Rating		4.6
Proposed Question:			

Plant conditions:

- The reactor is tripped following a spurious turbine trip.
- No Safety Injection.
- The crew has transitioned from E-0 to ES-0.1, "Reactor Trip Response".
- Process radiation monitors indicate as follows:
 - RM6482-1 Main Steam STM LN B in HIGH ALARM.
 - RM6511-1 Steam Generator Blowdown LOOP 2 in HIGH ALARM.

Based on these conditions what is the status of CSFST Radiation 'R' **(1)** and what mitigating strategy will be implemented **(2)** in response?

(1)	(2)
A. Yellow	Process OS1227.02 in parallel as the current conditions will not be addressed by the EOP network.
B. Yellow	Continue to process ES-0.1 and evaluate conditions for SI. Parallel use of OS1227.02 will interrupt timely execution of ES-0.1.
C. Orange	Process OS1227.02 in parallel as the current conditions will not be addressed by the EOP network.
D. Orange	Continue to process ES-0.1 and evaluate conditions for SI. Parallel use of OS1227.02 will interrupt timely execution of ES-0.1.

Proposed Answer:	A.	
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Explanation (Optional):

SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must demonstrate knowledge of the implementation and coordination of AOPs and EOPs.

A is correct. With the given process radiation monitors in high alarm, R will indicate yellow. The given conditions are consistent with a SGTL following a reactor trip. To mitigate this event, OS1227.02 will be processed in parallel with the EOP. This is a specific example of implementation of EOP/AOP as given in OP9.2

B is incorrect but plausible. Part (1) is correct. It is plausible that parallel use of OS1227.02 in this

<p>case will interrupt timely execution of ES-0.1. As stated in OP-9.2, "The use of other procedures in parallel (at the same time) with EOPs should generally be avoided as it could interrupt the timely execution of the emergency procedure in effect." OS1227.02 however is an exception to this and is given as an example in OP-9.2.</p> <p>C is incorrect but plausible. With the severity of a SGTL is it plausible that the R tree would indicate orange for the given conditions. Part (2) is correct.</p> <p>D is incorrect but plausible. With the severity of a SGTL is it plausible that the R tree would indicate orange for the given conditions. It is plausible that parallel use of OS1227.02 in this case will interrupt timely execution of ES-0.1. As stated in OP-9.2, "The use of other procedures in parallel (at the same time) with EOPs should generally be avoided as it could interrupt the timely execution of the emergency procedure in effect." OS1227.02 however is an exception to this and is given as an example in OP-9.2.</p>			
<p>Technical Reference(s):</p>		<p>OP 9.2, "Transient Response Procedure User's Guide" Rev 19, section 4.9.4 Concurrent Use of Procedures</p> <p>F-8, "RDMS (R)" Rev 20</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1195I 05</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		
	<p>Comprehension or Analysis</p>		<p>x</p>
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>		
	<p>55.43</p>	<p>(5)</p>	
<p>Comments:</p>			

Examination Outline Cross-reference:	Level	RO	SRO
Q91	Tier #		2
	Group #		2
	K/A #	034 (SF8 FHS) Fuel-Handling Equipment A2.01 Ability to (a) predict the impacts of the following malfunctions or operations on the Fuel Handling System ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Dropped fuel element	
	Importance Rating		4.4
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Core offload in progress. • The fuel handlers were moving an irradiated fuel assembly from the reactor core to the fuel transfer canal. • You are notified that the fuel assembly was accidentally dropped. • Manipulator Crane Radiation Monitors RM-6535A-1 and RM-6535B-1 have gone into alarm. <p>What actions are required?</p> <p>A. Enter procedure OS1215.06, "Fuel Handling Accident". Instruct the Refueling SRO to verify that the fuel assembly is located on the refueling cavity floor.</p> <p>B. Enter procedure OS1215.06, "Fuel Handling Accident". Evacuate non-essential personnel from the containment building.</p> <p>C. Enter procedure OS1215.02, "Area High Radiation". Evacuate non-essential personnel from the containment building.</p> <p>D. Enter procedure OS1215.02, "Area High Radiation". Notify the Shift Manager, HP, and Chemistry.</p>			

Proposed Answer:		B.	
Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem to select the correct AOP and determine what actions are required in response to the dropped fuel assembly.</p> <p>B is correct. Notification of a fuel handling accident is an entry condition into OS1215.06. The procedure directs evacuation of non-essential personnel and isolation of containment ventilation.</p> <p>A is incorrect but plausible. The correct procedure is entered. The procedure does direct isolation of containment building ventilation, however, the procedure directs immediate evacuation of non-essential personnel. A loss of refueling cavity level will direct placing fuel assembly on the cavity floor.</p> <p>D is incorrect but plausible. The manipulator crane is an area radiation monitor for refueling operations, however the given conditions in the question stem are such that OS1215.06, "Fuel Handling Accident" is the appropriate procedure.</p> <p>C is incorrect but plausible. The manipulator crane is an area radiation monitor for refueling operations, however the given conditions in the question stem are such that OS1215.06, "Fuel Handling Accident" is the appropriate procedure. Additionally, immediate evacuation of non-essential personnel is appropriate.</p>			
Technical Reference(s):		OS1215.06, "Fuel Handling Accident." Rev 16	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1192I 06		
Question Source:	Bank #	x	TEB 30034
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2007 Seabrook NRC Exam		

2020 Seabrook Station NRC Written Exam
 ES-401-5 Written Examination Question Worksheet

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO																				
Q92	Tier #		2																				
	Group #		2																				
	K/A #	072 (SF7 ARM) Area Radiation Monitoring 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:																							
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Radiation Monitor RM-6549, "Lo Range Spent Fuel Pool" is in high alarm. • Spent Fuel Pool Level is 25.5 feet. <p>What procedure will be entered (1) and what actions will be taken to minimize personnel exposure (2)?</p> <table border="0" style="width: 100%; border-collapse: collapse;"> <thead> <tr> <th style="width: 50%;"></th> <th style="width: 50%; text-align: center;">(1)</th> <th style="width: 50%;"></th> <th style="width: 50%; text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A.</td> <td>OS1252.03, "Area High Radiation".</td> <td></td> <td>Evacuate the Fuel Storage Building.</td> </tr> <tr> <td>B.</td> <td>OS1252.03, "Area High Radiation".</td> <td></td> <td>Place FAH in the Fuel Handling mode.</td> </tr> <tr> <td>C.</td> <td>OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".</td> <td></td> <td>Evacuate the Fuel Storage Building.</td> </tr> <tr> <td>D.</td> <td>OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".</td> <td></td> <td>Place FAH in the Fuel Handling mode.</td> </tr> </tbody> </table>					(1)		(2)	A.	OS1252.03, "Area High Radiation".		Evacuate the Fuel Storage Building.	B.	OS1252.03, "Area High Radiation".		Place FAH in the Fuel Handling mode.	C.	OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".		Evacuate the Fuel Storage Building.	D.	OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".		Place FAH in the Fuel Handling mode.
	(1)		(2)																				
A.	OS1252.03, "Area High Radiation".		Evacuate the Fuel Storage Building.																				
B.	OS1252.03, "Area High Radiation".		Place FAH in the Fuel Handling mode.																				
C.	OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".		Evacuate the Fuel Storage Building.																				
D.	OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".		Place FAH in the Fuel Handling mode.																				
Proposed Answer:	A.																						
Explanation (Optional):																							
SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem to select																							

the correct AOP and determine what actions are required to minimize personnel exposure.

A is correct. RM-6549 is an area radiation monitor. With this rad monitor in alarm, OS1252.03 is required. To minimize personnel exposure, the FSB will be evacuated.

B is incorrect but plausible. RM-6549 is an area radiation monitor. With this rad monitor in alarm, OS1252.03 is required. To minimize personnel exposure, the FSB will be evacuated although it is plausible the placing the fuel air handling system in the fuel handling mode would achieve this. The FAH mode is used to limit radiation release during a fuel handling accident.

C is incorrect but plausible. OS1215.07 contains a caution that “installed area monitors should be trended for changing radiological conditions in the fuel building” however, this is not an entry condition for OS1215.07. High radiation level on RM-6549 is a criterion for emergency action levels RU2 (Unusual Event) and RA2 (Alert). These emergency action levels are associated with a loss of spent fuel pool level or damage to irradiated fuel. High radiation is an alternate and sufficient condition to determine that spent fuel pool level has decreased when determining the applicability of the emergency action levels. If the student mistakenly applied the high radiation to selection of the AOP vs the emergency action level determination, this would be a plausible answer.

D is incorrect but plausible. OS1215.07 contains a caution that “installed area monitors should be trended for changing radiological conditions in the fuel building”. This is not an entry condition however. The FAH mode is used to limit radiation release during a fuel handling accident. High radiation level on RM-6549 is a criterion for emergency action levels RU2 (Unusual Event) and RA2 (Alert). These emergency action levels are associated with a loss of spent fuel pool level or damage to irradiated fuel. High radiation is an alternate and sufficient condition to determine that spent fuel pool level has decreased when determining the applicability of the emergency action levels. If the student mistakenly applied the high radiation to selection of the AOP vs the emergency action level determination, this would be a plausible answer.

Technical Reference(s):	OS1215.07, “Loss of Spent Fuel Pool Cooling or Level” Rev 19		
	OS1252.03, “Area High Radiation” Rev 15		
Proposed references to be provided to applicants during examination:	None		
Learning Objective:	SBK LOP L1187I 10		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			

2020 Seabrook Station NRC Written Exam
 ES-401-5 Written Examination Question Worksheet

Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q93	Tier #		2
	Group #		2
	K/A #	079 (SF8 SAS**) Station Air 2.4.6 Knowledge symptom based EOP mitigation strategies.	
	Importance Rating		4.7
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • Loss of Instrument Air has occurred. • The plant was tripped due to loss of plant control. • Following the plant trip, a red condition on the 'H' critical safety function occurs due to a loss of secondary heat sink. • The RCPs have been stopped. • There is no EFW flow. • The EFW pumps and SUFP cannot be started. • The crew is attempting to establish feed flow from the Condensate System. <p>How will the loss of Instrument Air impact the crew's ability to establish Condensate flow in FR.H-1, "Response to Loss of Secondary Heat Sink"?</p> <p>A. No impact, flow will be established through the EFW header.</p> <p>B. The preferred flow path is through the normal feed header. This will require restoration of IA</p> <p>C. The RCS cannot be depressurized as required because CS-V-185, "Pressurizer Aux Spray" cannot be opened. Flow cannot be established.</p> <p>D. SGs cannot be depressurized because the steam dump valves are failed closed on the loss of IA. Flow cannot be established.</p>			
Proposed Answer:	A.		

Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem and identify the correct procedural strategy to establish condensate flow in FR-H.1. Distracters B and D are partly correct from a systems only standpoint, however the candidate cannot correctly answer the question solely based on systems knowledge. They must possess knowledge of the detailed procedural strategy including alternate methods of performing required actions.</p> <p>A is correct. Step 7 of FR-H.1, "Response to Loss of Secondary Heat Sink" will establish condensate flow through the EFW header. The valves used are MOVs and a loss of IA will not impact this flow path.</p> <p>B is incorrect but plausible. If condensate flow cannot be established through the EFW header, flow will be established through the normal feed path. This flow path will include the normal feed reg or bypass valves which will require instrument air to open. This strategy is only implemented if the EFW flow path is not available. It is not the preferred flow path.</p> <p>C is incorrect but plausible. CS-V-185 will fail closed on a loss of containment instrument air. This is a separate air system. If this valve were closed the PORVs would be used to depressurize the SGs. The student must understand that the procedural strategy uses the PORVs if Aux spray is unavailable and that the PORVs are available with the loss of IA. This makes the statement 'flow cannot be established' incorrect.</p> <p>D is incorrect but plausible. The steam dumps valves will fail closed on a loss of IA. This is the preferred method for depressurizing the SG. If the steam dumps are unavailable, the ASDVs will be used to depressurize the SG to allow condensate flow. The student must understand that the procedural strategy uses the SDs as the preferred depressurization method and that the ASDVs are available as a backup means of depressurizing. This makes the statement 'flow cannot be established' incorrect.</p>			
Technical Reference(s):		FR-H.1, "Response to Loss of Secondary Heat Sink" Rev 37	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1211I 02		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	

2020 Seabrook Station NRC Written Exam
 ES-401-5 Written Examination Question Worksheet

Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:		Level	RO	SRO
Q94		Tier #		3
		Group #		
		K/A #	Conduct of Operations 2.1.29 Knowledge of how to conduct system lineups, such as valves, breakers, switches, etc.	
		Importance Rating		4.0
Proposed Question:				
<p>While completing an ODI.45A, "System Lineup and Review Exception Sheet" one component will not be realigned to the lineup position due to continued system operation and is being controlled by a procedure that is <u>not an MPE</u>.</p> <p>Per ODI.45, "System Lineup Performance", who must initial to verify that the controlling procedure will be complete and will realign the component?</p> <p>A. Any two SROs. B. Any one SRO. C. US only. D. SM only.</p>				
Proposed Answer:	A.			
Explanation (Optional):				
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(3), facility licensee procedures required to obtain authority for design and operating changes in the facility. Additionally, task SBK 1190102002 "REVIEW VALVE LINEUP SHEETS" is an SRO only task. This is SRO only knowledge.</p> <p>A is correct. From ODI.45, "System Lineup Performance", "If the method of configuration control is a procedure or procedure section other than an MPE than two independent SRO's shall verify that the procedure or procedure section will be performed in order to ensure positive configuration control."</p> <p>B is incorrect but plausible. If the procedure maintaining configuration control is an MPE, one SRO must initial the exception sheet.</p>				

C and D are incorrect but plausible. It is plausible and consistent with other requirements that a US or SM only are required for the given approval.			
Technical Reference(s):		ODI.45A Rev 11, page 2 of 2, item 'L'.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1305I 15		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	(3)	
Comments: Task SBK 1190102002 "REVIEW VALVE LINEUP SHEETS" is an SRO only task. This is SRO only knowledge.			

Examination Outline Cross-reference:	Level	RO	SRO										
Q95	Tier #		3										
	Group #												
	K/A #	Equipment Control 2.2.17 Knowledge of the process for managing maintenance activities during power operations, such as risk assessments, work prioritization, and coordination with the transmission system operator.											
	Importance Rating		3.8										
Proposed Question:													
<p>The plant is at 100% power. The US receives notification from ISO New England that Post Contingent Voltage is less than 345 kV.</p> <p>What procedure is required in response to this notification (1) and what is the Tech Spec implication (2)?</p> <table border="0" style="width: 100%;"> <thead> <tr> <th style="text-align: center; width: 40%;">(1)</th> <th style="text-align: center;">(2)</th> </tr> </thead> <tbody> <tr> <td>A. ON1246.03, "GSU Trouble"</td> <td>Perform Surveillance Requirement 4.8.1.1.1.a for the offsite power sources within 1 hour and at least once per 8 hours thereafter.</td> </tr> <tr> <td>B. ON1246.03, "GSU Trouble"</td> <td>Declare both offsite power sources inoperable and enter Tech Spec 3.8.1.1 action e.</td> </tr> <tr> <td>C. OS1246.02, "Degraded Vital AC Power (Plant Operating)"</td> <td>Perform Surveillance Requirement 4.8.1.1.1.a for the offsite power sources within 1 hour and at least once per 8 hours thereafter.</td> </tr> <tr> <td>D. OS1246.02, "Degraded Vital AC Power (Plant Operating)"</td> <td>Declare both offsite power sources inoperable and enter Tech Spec 3.8.1.1 action e.</td> </tr> </tbody> </table>				(1)	(2)	A. ON1246.03, "GSU Trouble"	Perform Surveillance Requirement 4.8.1.1.1.a for the offsite power sources within 1 hour and at least once per 8 hours thereafter.	B. ON1246.03, "GSU Trouble"	Declare both offsite power sources inoperable and enter Tech Spec 3.8.1.1 action e.	C. OS1246.02, "Degraded Vital AC Power (Plant Operating)"	Perform Surveillance Requirement 4.8.1.1.1.a for the offsite power sources within 1 hour and at least once per 8 hours thereafter.	D. OS1246.02, "Degraded Vital AC Power (Plant Operating)"	Declare both offsite power sources inoperable and enter Tech Spec 3.8.1.1 action e.
(1)	(2)												
A. ON1246.03, "GSU Trouble"	Perform Surveillance Requirement 4.8.1.1.1.a for the offsite power sources within 1 hour and at least once per 8 hours thereafter.												
B. ON1246.03, "GSU Trouble"	Declare both offsite power sources inoperable and enter Tech Spec 3.8.1.1 action e.												
C. OS1246.02, "Degraded Vital AC Power (Plant Operating)"	Perform Surveillance Requirement 4.8.1.1.1.a for the offsite power sources within 1 hour and at least once per 8 hours thereafter.												
D. OS1246.02, "Degraded Vital AC Power (Plant Operating)"	Declare both offsite power sources inoperable and enter Tech Spec 3.8.1.1 action e.												

Proposed Answer:		D.		
Explanation (Optional):				
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must select the correct AOP and the correct strategy which applies to the operability of the offsite power sources.</p> <p>D is correct. Notification by ISO New England that Post Contingent Voltage is less than 345 kV is an entry criteria for OS1246.02, "Degraded Vital AC Power (Plant Operating)". Upon entry to this procedure, step 17 will declare both offsite power sources inoperable and direct entry in TS 3.8.1.1 action e.</p> <p>A and B are incorrect but plausible. Notification by ISO New England that Post Contingent Voltage is less than 345 kV is similar to entry conditions in ON1246.03, "GSU Trouble". It is conceivable that degraded grid voltage would impact the GSUs and require AOP response. Both offsite power sources are inoperable with degraded grid voltage.</p> <p>C is incorrect but plausible. Notification by ISO New England that Post Contingent Voltage is less than 345 kV is an entry criteria for OS1246.02, "Degraded Vital AC Power (Plant Operating)". Upon entry to this procedure, step 17 will declare both offsite power sources inoperable and direct entry in TS 3.8.1.1 action e. Operability of offsite power sources is demonstrated by performing SR 4.8.1.1.1.a and b. This is performed if one offsite source is inoperable or if a diesel generator is inoperable. The required AOP does not allow for this however. The sources are declared inoperable, not demonstrated operable.</p>				
Technical Reference(s):		ON1246.03, "GSU Trouble" Rev 15 OS1246.02, "Degraded Vital AC Power (Plant Operating)" Rev 22		
Proposed references to be provided to applicants during examination:				None
Learning Objective:	SBK LOP L1199I 09, 10			
Question Source:	Bank #			
	Modified Bank#			(Note changes or attach Parent)
	New	x		

2020 Seabrook Station NRC Written Exam
 ES-401-5 Written Examination Question Worksheet

Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	x	
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q96	Tier #		3
	Group #		
	K/A #	Equipment Control 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.	
	Importance Rating		4.5
Proposed Question:			
<p>The plant is at 100% power. Engineering reports that rod H-8 is known to be untrippable. What is the required TS action and the required time for NRC notification for this condition?</p> <p>A. Determine that the shutdown margin requirement is satisfied within 1 hour and reduce thermal power to less than or equal to 75% within the next hour. Notify the NRC within 4 hours.</p> <p>B. Determine that the shutdown margin requirement is satisfied within 1 hour and reduce thermal power to less than or equal to 75% within the next hour. Notify the NRC within 24 hours.</p> <p>C. Determine that the shutdown margin requirement is satisfied within 1 hour and be in Hot Standby within 6 hours. Notify the NRC within 4 hours.</p> <p>D. Determine that the shutdown margin requirement is satisfied within 1 hour and be in Hot Standby within 6 hours. Notify the NRC within 24 hours.</p>			
Proposed Answer:	C.		
Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(2) Facility Operating Limitations in the Technical Specifications and Their Bases. The Technical Specifications component while related to the selected K/A is RO knowledge however, the responsibility and knowledge of NRC reports is specific to the SRO position.</p> <p>C is correct. Per TS 3.1.3.1.a a known untrippable control rod requires determination of SDM within 1 hour and to be in HSB within 6 hours. LI-AA-102-1001, "Regulatory Reporting" requires a 4-hour report for the initiation of any shutdown required by TS.</p> <p>A is incorrect but plausible. SDM determination is required however, reducing power to <75% is</p>			

required for a trippable but inoperable control rod not an untrippable rod.			
B is incorrect but plausible. SDM determination is required however, reducing power to <75% is required for a trippable but inoperable control rod. Additionally, the NRC notification is required within 4 hours, not 24. 24 hours is the threshold for determination of immediate reportability by operations.			
D is incorrect but plausible. Per TS 3.1.3.1.a a known untrippable control rod requires determination of SDM within 1 hour and to be in HSB within 6 hours. LI-AA-102-1001, "Regulatory Reporting" requires a 4-hour report for the initiation of any shutdown required by TS, not a 24-hour report. 24 hours is the threshold for determination of immediate reportability by operations.			
Technical Reference(s):		Technical Specifications, 3.1.3.1 Rev 141. LI-AA-102-1001, "Regulatory Reporting" Rev 28.	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8031I 23 SBK LOP L1305I 09		
Question Source:	Bank #	x	12402
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:	2012 Turkey Point NRC Exam – SRO Question		
Question Cognitive Level:	Memory or Fundamental Knowledge	x	
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43	(2)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q97	Tier #		3
	Group #		
	K/A #	Radiation Control 2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.	
	Importance Rating		3.7
Proposed Question:			
<p>Plant conditions:</p> <ul style="list-style-type: none"> • A LOCA outside containment occurred at 0130. • A Site Area Emergency was declared at 0140. • The broken line was manually isolated locally, but the operator performing the task was injured and cannot leave the area on his own. • Initial dose rate estimates are 110 R/hr gamma. • The rescue time for a 2-man team is estimated to be 10 minutes with a maximum of 15 minutes. <p>Under these circumstances, a rescue attempt _____</p> <p>A. by risk-informed volunteers may proceed ONLY with Site Emergency Director authorization.</p> <p>B. by risk-informed volunteers may proceed ONLY with Radiological Controls Supervisor authorization.</p> <p>C. may be made by qualified individuals selected and approved by the Radiological Controls Coordinator.</p> <p>D. may be made without special authorization since 10CFR20 exposure limits will NOT be exceeded.</p>			
Proposed Answer:	A.		
Explanation (Optional):			
SRO justification: This question meets SRO level screening criteria 10CFR55.43(b)(4), Radiation hazards that may arise during normal and abnormal situations, including maintenance activities			

and various contamination conditions. Specifically, the Emergency Dose limits that are allowed to perform lifesaving activities.			
A is correct. Given the conditions in the question stem, the rescue team will be performing a "lifesaving activity". The dose for each member of the rescue team will be $(110\text{R/hr})(.25\text{hr})=27.5\text{ R}$. Per procedure ER-4.3, "Radiation Protection During Emergency Conditions", "Figure 2: Emergency Dose Limits", a person may receive a dose of $>25\text{R}$ for the purpose of performing a lifesaving activity or protecting large populations. The dose is allowed "only on a voluntary basis to persons fully aware of the risks involved". This Emergency Dose Limit allowance requires STED or SED approval.			
B. Incorrect but plausible. It is plausible during an emergency the Rad Con Supervisor has the authority to approve emergency dose limits.			
C. Incorrect but plausible. It is true that the rescue attempt may be performed, however the emergency dose extension must be approved by the STED or SED.			
D. Incorrect but plausible. This distractor is plausible if the student misinterprets the conditions in the question stem, or has false knowledge of the emergency dose limit criteria.			
Technical Reference(s):		ER 4.3, Radiation Protection During Emergency Conditions, Rev 33	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1525I 15		
Question Source:	Bank #	x	100892
	Modified Bank#		(Note changes or attach Parent)
	New		
Question History:		Seabrook 2015 NRC Exam (Question used on one of the two previous NRC exams)	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		

2020 Seabrook Station NRC Written Exam
ES-401-5 Written Examination Question Worksheet

	55.43	(4)
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
Q98	Tier #		3
	Group #		
	K/A #	Radiation Control 2.3.7 Ability to comply with radiation work permit requirements during normal or abnormal conditions.	
	Importance Rating		3.6
Proposed Question:			
<p>Per Technical Specification section 6.11, "High Radiation Area", what are the minimum required controls for areas accessible to personnel with radiation levels >1000 mR/hour?</p> <p>A. Locked doors to prevent unauthorized entry. Keys maintained under the administrative control of RP technicians.</p> <p>B. Locked doors to prevent unauthorized entry. Keys maintained under the administrative control of the SM and/or RP supervision.</p> <p>C. Locked doors <u>and</u> remote continuous surveillance to prevent unauthorized entry. Keys maintained under the administrative control of RP technicians.</p> <p>D. Locked doors <u>and</u> remote continuous surveillance to prevent unauthorized entry. Keys maintained under the administrative control of the SM and/or RP supervision.</p>			
Proposed Answer:	B.		
Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(4) Radiation Hazards That May Arise during Normal and Abnormal Situations, including Maintenance Activities and Various Contamination Conditions. The candidate must recognize the conditions given as applying to the locked high radiation area and recall how access is controlled. This question cannot be answered solely based on RO knowledge of radiological safety principles.</p> <p>B is correct. TS 6.11 describes two High Radiation areas, based on dose rate. The student must recognize that radiation levels >1000 mR/hour require posting as a <u>Locked</u> High Radiation area. Then they must recall what the access requirements are in accordance with the administrative section of TS.</p>			

A is incorrect but plausible. Keys for locked high radiation areas must be controlled by the SM or RP supervision, not RP technicians.			
C and D are incorrect but plausible			
Remote continuous surveillance is allowed in lieu of the RWP stay time. It is not required in addition to locked doors.			
Technical Reference(s):		Technical Specifications 6.11 Rev 141	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L1307I 04		
Question Source:	Bank #	<input checked="" type="checkbox"/>	TEB 31609
	Modified Bank#	<input type="checkbox"/>	(Note changes or attach Parent)
	New	<input type="checkbox"/>	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge	<input checked="" type="checkbox"/>	
	Comprehension or Analysis	<input type="checkbox"/>	
10 CFR Part 55 Content:	55.41		
	55.43	(4)	
Comments:			

Examination Outline Cross-reference:	Level	RO	SRO
Q99	Tier #		3
	Group #		
	K/A #	Emergency Procedures/Plan 2.4.17 Knowledge of EOP terms and definitions.	
	Importance Rating		4.3
Proposed Question:			
<p>In accordance with OP9.2, "Transient Response Procedure User's Guide" if containment pressure exceeds 4 psig during an accident _____</p> <p>A. only post-accident monitoring (PAM) indications may be used.</p> <p>B. control room indications must be verified using redundant indications.</p> <p>C. adverse containment parameters must be used for the duration of the accident.</p> <p>D. adverse containment parameters must be used until containment pressure decreases to below 4 psig.</p>			
Proposed Answer:	D.		
Explanation (Optional):			
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must have knowledge of administrative procedure OP9.2 and understand the requirements for use of adverse condition parameters and correctly implement EOPs.</p> <p>D is correct. In those instances in the EOPs where both the normal and adverse containment process parameter values are given, the operator decides which of the two values to use by determining the containment pressure and radiation conditions. If containment pressure exceeds 4 psig, the operator would implement the procedures using the adverse containment (post-accident) process parameter values. Alternately, if containment pressure is less than approximately 4 psig, the operator would use the normal containment.</p> <p>A is incorrect but plausible. The post-accident monitoring indications are a system of indications that are relied on to perform their design function in a harsh environment following an accident in containment. Use of the PAM indications is not required by OP9.2.</p>			

<p>B is incorrect but plausible. It is good practice to verify parameters using redundant and diverse indications however; this is not required by OP9.2.</p> <p>C is incorrect but plausible. Use of adverse containment parameters is suspended when containment pressure is below 4 psig. It is conceivable that their use would be required for the duration of an accident.</p>			
<p>Technical Reference(s):</p>		<p>OP 9.2, "Emergency Operators Users Guide" Rev 19</p>	
<p>Proposed references to be provided to applicants during examination:</p>			<p>None</p>
<p>Learning Objective:</p>	<p>SBK LOP L1195I05</p>		
<p>Question Source:</p>	<p>Bank #</p>		
	<p>Modified Bank#</p>		<p>(Note changes or attach Parent)</p>
	<p>New</p>	<p>x</p>	
<p>Question History:</p>			
<p>Question Cognitive Level:</p>	<p>Memory or Fundamental Knowledge</p>		<p>x</p>
	<p>Comprehension or Analysis</p>		
<p>10 CFR Part 55 Content:</p>	<p>55.41</p>		
	<p>55.43</p>	<p>(5)</p>	
<p>K/A/ match justification: SROs are responsible for implementing procedures at Seabrook. Hence it is the SROs responsibility to understand and correctly apply adverse containment parameters.</p>			

Examination Outline Cross-reference:		Level	RO	SRO
Q100		Tier #		3
		Group #		
		K/A #	Emergency Procedures/Plan 2.4.35 Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.	
		Importance Rating		4.0
Proposed Question:				
<p>Plant conditions:</p> <ul style="list-style-type: none"> • The crew is performing step 5 of ECA-0.0, "Loss of All AC Power". • 4160V Bus 6 is locked out due to a ground fault. • 'A' EDG has been started from the control room and its output breaker is closed. • 'A' train EPS has failed it is not sequencing. <p>What actions are required in response to the failure of EPS?</p> <p>A. Reset RMO and manually start equipment as necessary. B. Place RMO bypass switch in bypass and manually start equipment as necessary. C. Perform Attachment 'B' to deactivate EPS locally. When EPS is deactivated, perform step 6. D. Go to step 6. When step 6 has been completed, perform Attachment 'B' to deactivate EPS locally.</p>				
Proposed Answer:	D.			
Explanation (Optional):				
<p>SRO justification: This question meets SRO only criteria for 10CFR5543(b)(5), assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations. The candidate must assess plant conditions as given in the stem to select the correct strategy and coordinate implementation of local actions via attachment with procedure steps.</p>				

<p>D is correct. Per ECA-0.0, "Loss of All AC Power", with the EDG output breaker closed but EPS failed, step 6 must first be performed before de energizing EPS locally.</p> <p>A is incorrect but plausible. If an individual component has failed to start once sequenced by EPS, RMO will be reset and the equipment will be manually started.</p> <p>B is incorrect but plausible. The RMO bypass switch is used to bypass the RMO feature to close a UAT or RAT breaker once offsite power is restored. It is not used to bypass a failed EPS in this case.</p> <p>C is incorrect but plausible. Per ECA-0.0, "Loss of All AC Power", with the EDG output breaker closed but EPS failed, step 6 must first be performed before de energizing EPS locally. The student must understand coordination of step 6 with Attachment 'B' in order to correctly respond to the failure of EPS.</p>			
Technical Reference(s):		ECA-0.0, "Loss of All AC Power" Rev 55	
Proposed references to be provided to applicants during examination:			None
Learning Objective:	SBK LOP L8067I 03		
Question Source:	Bank #		
	Modified Bank#		(Note changes or attach Parent)
	New	x	
Question History:			
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		x
10 CFR Part 55 Content:	55.41		
	55.43	(5)	
Comments:			