U.S. Nuclear Regulatory Commission

Site-Specific RO Written Examination

Applicant Information

Name:	
Date:	Facility/Unit: SEABROOK STATION
Region: I ☒ II ☐ III ☐ IV ☐	Reactor Type: W ☒ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination, you must achieve a final grade of at least 80.00 percent. Examination papers will be collected 6 hours after the examination begins.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

Examination Value	75	Points
Applicant's Score	______	Points
Applicant's Grade	______	Percent

Question Number 1

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	1	
Group #	1	
K/A #	008AK2.02	
Importance Rating	2.7	2.7

KA 008 = Pressurizer vapor space accident
AK2. Knowledge of the interrelations between the Pressurizer Vapor Space Accident and the following: AK2.02 Sensors and detectors

Proposed Question:

The following plant conditions exist:

- A Pressurizer Power Operated Relief Valve (PORV) has failed open.
- The PORV cannot be closed or isolated.
- A Safety Injection Signal has been actuated.
- RCS Pressure is stable at 1280 psig
- Pressurizer Relief Tank pressure is 35 psig
- Pressurizer Relief Tank temperature is 140°F

What will the PORV Tailpipe Temperature Indicator read?

A. 251°F

B. 281°F

C. 298°F

D. 577°F

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The saturation temperature for 35 psig (50 psia) is 280.55 degrees.

Answer A is incorrect. 251 degrees is the saturation temperature for atmospheric pressure. Students that made a mistake converting PSIG to PSIA may have chosen this answer.

Answer C is incorrect. 298 degrees is between the saturation temperature for 35 psig and 1280 psig. If the students assumed that the steam in the PORV tailpipe was undergoing an adiabatic throttling then this answer could be chosen.

Answer D is incorrect. 577 degrees is the saturation temperature for 1280 psig. Students that assume the steam in the PORV tailpipe does not change conditions until the steam enters the Pressurizer Relief tank may choose this answer.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Steam Tables.

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: L1413I07RO (As available)

Question Source: Bank # 22202
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	011EK1.01	
	Importance Rating	4.1	4.4

K/A 011 = Large Break LOCA
EK1 Knowledge of the operational implications of the following concepts as they apply to the Large Break LOCA : EK1.01 Natural circulation and cooling, including reflux boiling

Proposed Question

What is the primary method of decay heat removal for large break LOCA's?

A. The condensation of reflux boiling in the S/Gs.

B. Heat transfer between the RCS and the S/Gs due to forced circulation flow.

C. Heat transfer between the RCS and the S/Gs due to natural circulation flow.

D. The injection of water from the ECCS and leakage of steam/water out the break.

Proposed Answer: D

Explanation (Optional):

D is correct. The background document for E-1 discusses the injection of water from the RWST and the removal of steam/water out the break as the primary method of heat removal for a large break LOCA .

A is incorrect. Reflux cooling is a mechanism for cooling during a small break LOCA before the RCS has drained enough to clear the loop seal.

B is incorrect. For smaller break sizes forced circulation is a viable method of heat removal during the initial phase of the transient but the primary side of the Steam generators are drained relatively soon in a large break LOCA.

C is incorrect. Natural circulation also provides a viable method of heat removal post trip and even for the initial phase of a small break LOCA but the primary side of the Steam generators are drained relatively soon in a large break LOCA.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse background documents for E-1, Section 2.1, Description of Loss of Reactor Coolant Accidents.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1413I 03 (As available)

Question Source: Bank # 14285
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.8 /41.10
55.43 ______

Comments:

Question 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	017 G 2.2.39	
	Importance Rating	3.9	4.5

K/A 017 = G APE, RCP Malfunctions, loss of RC flow
2.2 Equipment Control: 2.2.39 Knowledge of less than or equal to one hour Technical Specification action statements for systems.

Proposed Question:

The following conditions exist:

- The plant is in Mode 4
- Reactor Coolant System Loop 3 is the only OPERABLE RCS loop, with the "C" RCP running.
- Train "A" RHR is in the shutdown cooling mode.
- Train "B" RHR is aligned in the ECCS mode.
- A problem during Design Modifications in the switchyard has caused a Loss of Power deenergizing Bus 1, 2, 3, 4, 5, and 6.
- Both Emergency Diesels have functioned as designed.

AT THE END OF THE EVENT which of the following correctly describes the actions required by technical specifications?

A. Immediately initiate actions to return a RCS loop to OPERABLE status and, if a RCS loop is not returned, be in COLD SHUTDOWN within 24 hours.

B. Immediately initiate actions to align the second RHR loop to the shutdown cooling mode and, if the second RHR loop is not aligned, be in COLD SHUTDOWN within 24 hours.

C. Immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System while immediately initiating actions to return a RCS loop to OPERABLE status.

D. No operations are permitted that would cause dilution of the Reactor Coolant System boron concentration and core outlet temperature must be maintained at least 10 degrees below saturation temperature.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. "Technical Specification 3.4.1.3 requires 2 loops OPERABLE and at least one of those loops in Operation. Initially The "C" RCP and associated SG were OPERABLE and the "A" RHR loop was OPERABLE in the shutdown cooling mode. The Loss of Power to all buses caused the "C" RCP to trip. After both EDGs sequenced their respective loads 1 cooling loop, the "A" RHR loop, was returned to OPERABLE in the SD cooling Mode. If less than 2 loops are OPERABLE Action "a." of TS 3.4.1.3 requires the immediate return of two loops to OPERABLE. Placing train "B" RHR in SD cooling is not an option because in Mode 4 it must remain aligned for ECCS. If the crew is able to re-energize bus 1 or 2 and restart an RCP then they would return to compliance. Additionally, because the only remaining loop is an RHR loop, the plant must be placed in COLD SHUTDOWN within 24 hours.

Answer B is incorrect. In mode 4 the B RHR loop must stay aligned for ECCS.

Answer C is incorrect. The distractor restates actions to be taken if no loop is in operation however the "A" RHR pump would be started by the Emergency Power Sequencer and would have already been aligned to the shutdown cooling mode. This would leave only the "A" RHR loop OPERABLE and in operation, so actions would be required to either restart an RCP or be in cold shutdown within 24 hrs.

Answer D is incorrect. This distractor restates the footnoted allowance to turn off all RCPs and all RHR pumps for up to 1 hour as required for maintenance or plant reconfigurations. This exception could not be applied to this case because the event was a casualty.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
T.S. 3.4.1.3, Reactor Coolant Loops and Coolant Circulation.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8021I 14RO (As available)

Question Source: Bank #
Modified Bank # 3361 (Note changes or attach parent)
New

Original TEB 3361

The following conditions exist:

- The plant is in Mode 4
- Reactor Coolant System Loop 3 is the only OPERABLE RCS loop, with the "C" RCP running.
- Due to common mode failure, both RHR loops are declared inoperable.

Which of the following correctly describes the actions required by technical specifications?

A. Immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

B. Within one hour, restore another loop to OPERABLE status, or be in COLD SHUTDOWN within 24 hours.

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

D. Be in COLD SHUTDOWN within 24 hours.

Proposed Answer: C

Question History:
Last NRC Exam: Original question never used on a NRC Exam
Exam Bank History: Modifed from bank
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7 /41.10
55.43 43.2

Comments:

Question 4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	022AK1.03	
	Importance Rating	3.0	3.4

K/A 022 = Generic APE, Loss of Reactor Coolant Makeup
AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of Reactor Coolant Makeup: AK1.03 Relationship between charging flow and PZR Level

Proposed Question:

The following plant conditions exist:

- The Plant is at 100% power
- All control systems are operating in AUTOMATIC
- Pressurizer level is 62% and STABLE
- VCT level is 50% and STABLE
- Charging system flow is 97 GPM
- Letdown flow is 80 GPM
- VCT level transmitter, CS-LT-112, fails LOW

Assuming NO operator action, how does this failure affect the plant?

A. Automatic Makeup is disabled. Charging pump suction will swap to the RWST. Charging flow will decrease to maintain Pressurizer level STABLE.

B. Automatic Makeup initiates. VCT divert is disabled. Charging system flow increases when the VCT fills causing Pressurizer level to increase. The plant will eventually trip on HI PRESSURIZER LEVEL.

C. Automatic Makeup initiates. Makeup will not automatically terminate. CS-LCV-112A/LV-112A Letdown divert valves will open to divert letdown to the Primary Drain tank/Boron Waste Storage Tanks respectively. Pressurizer level is unaffected.

D. Automatic Makeup is disabled. Charging pump swap over to the RWST is disabled. As VCT level decreases, Charging pumps will lose suction. Pressurizer level will decrease until charging flow is restored.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. CS-LT-112 provides the input for automatic makeup and when low level is sensed an automatic makeup initiates. With CS-LT-112 failed low the makeup will not terminate. CS-LT-185 will cause CS-LCV-112A/LV-112A Letdown divert valves to open based on actual level which will divert letdown to the Primary Drain tank/Boron Waste Storage Tanks respectively.

Answer A is incorrect. Automatic Makeup is only actuated from CS-LT-112 so it is not disabled but it would actuate erroneously. Both CS-LT-112 and LT-185 low level signals are required to swap Charging pump suction to the RWST. Charging flow is unaffected provided that ample level remains in the VCT.

Answer B is incorrect. Automatic Makeup would initiate, but VCT divert is not disabled. If the VCT went to 100% level charging system flow would likely be affected even with the level control system attempting to decrease charging flow.

Answer D is incorrect. Automatic Makeup is only actuated from CS-LT-112 so it is not disabled but it would actuate erroneously. Both CS-LT-112 and LT-185 low level signals are required to swap Charging pump suction to the RWST. VCT level does not decrease, so the Charging pumps will not lose suction.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8024I, Chemical Volume and Control System.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8024I 03 RO (As available)

Question Source: Bank #
Modified Bank # 22592 (Note changes or attach parent)
New

Original TEB 22592 :

The following plant conditions exist:

- The Plant is at 100% power
- All control systems are operating in AUTOMATIC
- VCT level is 44% and STABLE
- Letdown flow is 80 GPM
- VCT level transmitter, CS-LT-112, fails LOW

Assuming NO operator action, how does this failure affect the plant?

A. Automatic Makeup is disabled. Charging pump suction will swap to the RWST.

B. Automatic Makeup initiates. Makeup will be terminated by VCT high level at 90%.

C. Automatic Makeup initiates. Makeup will not automatically terminate. CS-LCV-112A/LV-112A Letdown divert valves will open to divert letdown to the Primary Drain tank/Boron Waste Storage Tanks respectively.

D. Automatic Makeup is disabled. Charging pump swap over to the RWST is disabled. As VCT level decreases, Charging pumps will lose suction.

Proposed Answer: C

Question History:
Last NRC Exam Original question never used on a NRC Exam
Exam Bank History Modified from bank
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.8 /41.10
55.43 ____

Comments:

Question 5

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	026AK3.02	
	Importance Rating	3.6	3.9

K/A 026 = Loss of Component Cooling Water
AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Component Cooling Water: AK3.02 The automatic actions (alignments) within the CCWS resulting from the actuation of the ESFAS

Proposed Question:

Given the following plant conditions:

- The Plant is operating at 25% power.
- While performing Solid State Protection System (SSPS Train A) testing, a Phase "B" Containment Isolation signal is generated and all associated valves close.
- The Phase "B" signal cannot be reset.

Based on these conditions, which of the following describes the required operator actions?

A. Initiate a MANUAL reactor trip and stop all RCPs within 10 minutes.

B. Trip the main turbine, insert control rods to achieve < 5% power, open reactor trip breakers, stop the A and D RCPs within 10 minutes.

C. Initiate a MANUAL reactor trip and stop the A and D RCPs. Three (3) to five (5) minutes later close all seal water return leakoff valves.

D. Restore Component Cooling Water to the thermal barrier heat exchangers within ten (10) minutes or initiate a MANUAL reactor trip and stop all RCPs.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. Each containment penetration has both a Train A and a Train B isolation valve. A phase "B" signal on either train will therefore isolate all PCCW cooling to ALL pumps. A MANUAL reactor trip followed by stopping all RCPs will be required within 10 minutes of the loss of PCCW cooling.

Answer B is incorrect. Students could mistakenly determined that PCCW cooling was only lost to the Reactor coolant pumps cooled by train "A" PCCW and action would be required to secure the reactor before stopping the A and D RCPs.

Answer C is incorrect. In this case students could mistakenly determine that PCCW cooling was only lost to the Reactor coolant pumps cooled by train "A" PCCW but no actions were allowed to secure the reactor before stopping A and D RCPs. The action to close seal leakoff valves was added to this distractor because the RCP malfunction abnormal requires that seal water return leakoff valves be closed three (3) to five (5) minutes after securing RCPs when the initiating event is failed RCP seals.

Answer D is incorrect. The Primary Component Cooling Water to the thermal barrier heat exchangers is not isolated with a Phase "B" signal so this cooling medium would not be lost. The requirement to restore the normal PCCW supply within ten (10) minutes or initiate a MANUAL reactor trip and stop all RCPs is correct but, would only be applied for an instance when NORMAL PCCW cooling is lost.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8036I Primary Component Cooling water.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8035I12 and L1445I01RO /L1445I03RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New (X)

Question History:
Last NRC Exam N/A, new question
Exam Bank History N/A, new question

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

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

10 CFR Part 55 Content: 55.41 41.5, 41.10
55.43 ____

Comments:

Question 6

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	027AA2.15	
	Importance Rating	3.7	4.0

K/A 027 = Pressurizer Pressure Control System Malfunction
AA2. Ability to determine and interpret the following as they apply to the Pressurizer Pressure Control Malfunctions: AA2.15 Actions to be taken if PZR pressure instrument fails high

Proposed Question

The following plant conditions exist:

- The plant is at 50% power with all control systems in AUTOMATIC.
- The controlling Pressurizer Pressure Channel slowly fails high.
- The Reactor Operator reports RCS pressure is 1940 psig and slowly decreasing.
- The Unit Supervisor enters OS1201.06, "PZR Pressure Instrument/Component Failure".
- Attempts to close RC-PCV-455A (PZR Spray Valve) are unsuccessful.

Which of the following actions should the crew perform next?

A. Commence a power reduction, and raise charging flow to compress the PZR bubble.

B. Trip the reactor, initiate safety injection, and enter E-0, "Reactor Trip or Safety Injection", while concurrently tripping the "A" RCP.

C. Energize ALL PZR heaters, commence a rapid power reduction in anticipation of tripping the reactor and stopping ALL Reactor Coolant Pumps.

D. Trip the reactor, when immediate actions of E-0, "Reactor Trip or Safety Injection", are complete, stop the "C" RCP and up to two more RCPs as necessary to stop RCS pressure drop.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. A failure of the controlling pressurizer channel high will cause spray valves to open and RCS pressure to drop. According to OS1206.01, if spray control has failed and PZR pressure drops to < 1945 psig, the reactor should be tripped and after Immediate Actions, the "C" RCP should be stopped. Based on Salem OE, the abnormal has been changed to reflect the fact that additional RCPs may need to be secured to stop the RCS pressure drop.

Answer A is incorrect because although it might be plausible to reduce power, it is not procedurally driven and because pressure is so low would be more conservative to trip the reactor in accordance with OS1206.01.

Answer B is incorrect because although tripping the reactor is correct and recently Seabrook has aligned ourselves with the industry with regard to initiating SI after tripping the reactor (vice tripping the reactor by initiating SI), in this case, SI is not required until <1800 psig, so this action would be incorrect. Additionally, the "A" RCP is associated with the "B" Spray Valve (human factoring)

Answer C is incorrect because this is non-procedurally driven and non-conservative in this situation. Additionally, stopping all RCPs is not desirable or correct as forced circulation is better than natural circulation.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1201.06, step 2

Proposed references to be provided to applicants during examination: None

Learning Objective: L1182I05RO (As available)

Question Source: Bank # 26984
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Used on 2005 NRC Exam
Exam Bank History Directly from bank
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 7

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	029 G 2.1.20	
	Importance Rating	4.6	4.6

029 = Anticipated Transient without Scram
2.1 Conduct of Operations
2.1.20 Ability to interpret and execute procedure steps.

Proposed Question:

The following plant conditions exist:

- An ATWS has occurred and actions are being taken in accordance with FR-S.1 "Response to Nuclear Power Generation/ATWS".
- Boration flow was initiated by starting a boric acid transfer pump and positioning CS-V426, EMERGENCY BORATION TO CHG PUMP SUCT HDR, to OPEN.
- PZR pressure is 2390 psig
- Charging flow (FI-121) is 45 gpm
- Boration flow (FI-183) is 82 gpm.
- VCT level is increasing.

What action should the operator take to INCREASE the boration rate?

A. Start an additional Boric Acid Pump.

B. Close CS-LCV-112B&C, the VCT outlet isolation valves

C. Start CS-P-128, Positive Displacement Pump and Charge at max rate.

D. Open PZR PORV(s) and block valves until PZR pressure is less than 2185 psig.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. RCS pressure is too high for effective emergency boration flow. FR-S.1 lowers RCS pressure by opening PZR PORV(s) and block valves until PZR pressure is less than 2185 psig.

Answer A is incorrect. If the boration flow was low because the first Boric acid pump had failed then starting an additional Boric Acid Pump would be effective but students should be able to determine the first boric acid pump is delivering 82 gpm that is going directly to the VCT.

Answer B is incorrect. Closing CS-LCV-112B&C, the VCT outlet isolation valves would be part of shifting the CS pumps from the VCT to the RWST but, even if completed, this would not increase the flow from the centrifugal charging pumps.

Answer C is incorrect. Starting CS-P-128, Positive Displacement Pump and Charge at max rate would provide flow even at this high pressure, but it is more desirable to lower RCS pressure and allow the higher flow centrifugal pumps to provide the boration flow.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
FR-S.1 "Response to Nuclear Power Generation/ATWS", Step 3.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1200I01RO (As available)

Question Source:	Bank #	20672	
	Modified Bank #		(Note changes or attach parent)
	New		

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Directly from bank

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

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

10 CFR Part 55 Content:	55.41	41.10 /45.12
	55.43	43.5

Comments:

Question 8

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	038EA1.04	
	Importance Rating	4.3	4.1

K/A 038 = Steam Generator Tube Rupture
EA1 Ability to operate and monitor the following as they apply to a SGTR: EA1.04 PZR spray, to reduce coolant system pressure

Proposed Question:

The following plant conditions exist:

- A Steam Generator Tube Rupture has occurred.
- All ECCS systems actuated properly.
- EFW flow has been isolated to the Ruptured SG and throttled to maintain 6% to 50% in the remaining intact SGs.
- The crew has determined their required Core Exit Thermocouple cooldown temperature.
- Before starting the cooldown the following RCS conditions are noted:
    - RCS sub-cooling is 100 °F
    - PZR level is 11% and SLOWLY INCREASING
    - RCS pressure is 2000 PSIG and SLOWLY INCREASING

Which of the following is the PREFERRED method to INITIALLY DEPRESSURIZE the RCS when the cooldown is started?

A. Cycle Pressurizer Heaters as necessary to maintain RCS Subcooling greater than 40 °F while cooling down to the target temperature. Sufficient indications of adequate RCS coolant invertory is maintained.

B. Use Auxiliary Spray to slowly depressurize the RCS to prevent a High Steam Pressure rate Steamline isolation. This will then require lowering Ruptured SG pressure by dumping steam to the atmosphere.

C. Use Normal Pressurizer Spray to depressurize the RCS to less than 1925 PSIG to allow blocking of the Low Steamline Pressure SI. This prevents an auto MSI and preserves the flowpath from the intact SGs to the Condenser.

D. Cycle a Pressurizer PORV to minmize break flow while helping to remove mass from the Pressurizer and minimize potential pressure excursions to the PZR code safety valves.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. E-3 uses Normal Pressurizer Spray to depressurize the RCS as the prefered method. The E-3 background document states that blocking the low steam line pressure SI preserves the flowpath from the intact SGs to the Condenser to minimize radiological consequences. Additionally if the cooldown had to be performed with the ASDVs this would increase the time required, thereby increasing the volume of leakage allowed to pass from primary to secondary.

Answer A is incorrect. Direction is given in E-3 to use Pressurizer Heaters to control RCS pressure as necessary. The initial RCS cooldown in E-3 is designed to ensure RCS subcooling is maintained. This direction does not provide for lowering RCS pressure as required to block the low steam line pressure SI.

Answer B is incorrect. Although Auxiliary Spray could be used to lower RCS pressure, E-3 uses Normal Spray as the first choice and a PORV if that is not available. After the Low Steamline SI is blocked a High Steam Pressure rate Steamline isolation can occur if the crew dumps steam too fast. If this isolation did occur the cooldown will be slower because the crew will now have to dump steam to the atmosphere.

Answer D is incorrect. The PORVs are the second choice for preforming the depressurization. The availablity of at least one PORV and block valve is verified in step 9 and the E-3 background document states that is to relieve RCS pressure excursions so the Code Safeties are not challenged.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse background document for E-3, concerning the note prior to step 7.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1205I 03RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A, new question
Exam Bank History N/A, new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 9

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	040AK1.07	
	Importance Rating	3.4	4.2

K/A 040 = Steam Line Rupture
AK1. Knowledge of the operational implications of the following concepts as they apply to Steam Line Rupture: AK1.07 Effects of feedwater introduction on dry S/G

Proposed Question:

The following plant conditions exist:

- The crew is responding to four faulted Steam Generators.
- Narrow Range level in all generators is off scale low.

Why should a MINIMUM of 25 gpm feed flow be maintained to each Steam Generator?

A. Minimize the RCS Cooldown.

B. Prevent thermal shock of Steam Generator components.

C. Prevent the water in the feed ring from flashing to steam.

D. Minimize the chance of potential loss of Secondary Heat Sink.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The background document for ECA-2.1 discusses that some minimal amount of flow should be maintained to prevent thermal shock of Steam Generator components when flow in re-introduced.

Answer A is incorrect. Lowering flow to 25 gpm does help to minimize the RCS cooldown, but a cooldown in this condition is still severe enough to present Cold Overpressure integrity challenges and can also result in re-criticality.

Answer C is incorrect. The J tube feed nozzles keep the feed ring full to prevent flashing.

Answer D is incorrect. The 25 gpm feed is only provided to Steam Generators that are below 6% (15% for Adverse Containment) NR level. The minimum level is to ensure some secondary Heat Sink remains available.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse background documents for ECA 2.1, caution prior to step 2.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1207I 05RO and L1207I06 RO (As available)

Question Source: Bank # 20703
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.8 /41.10
55.43 ______

Comments:

Question 10

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	054 G 2.2.44	
	Importance Rating	4.2	4.4

K/A 054 = Loss of Main Feedwater
2.2 Equipment Control:
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.

Proposed Question:

The following plant conditions exist:

- The plant is operating at 100% Power.
- S/G "A" Main Feed Regulating valve is in manual.
- S/G water levels are stable.
- PT-508 main feed header pressure transmitter fails low.

Assuming NO operator action, what is the effect on S/G feed pumps (SGFPs) and what automatic actions will take place to protect the plant?

A. Feed pumps speed-up due to pressure mismatch. Reactor trips on main turbine trip due to S/G High-High level.

B. Feed pumps slow down due to pressure mismatch. Decreasing S/G water levels cause reactor to trip on S/G level low-low.

C. Feed pumps slow down due to pressure mismatch. Main turbine trips on S/G low-low level. Reactor trips on main turbine trip.

D. Feed pumps speed-up due to pressure mismatch. Feed pumps trip on overspeed. Main turbine trips on loss of feed. Reactor trips on turbine trip.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. When PT-508 fails low the master speed controller will increase the Feed pumps speed due to pressure mismatch. With no operator intervention the 3 other feed regulating valves will close down but, with "A" FRV in manual, "A" SG level will increase to the high level trip setpoint. The Reactor trips on main turbine trip due to S/G High-High level.

Answer B is incorrect. The feed pumps will increase speed vice slowing down. If the feed pumps did slow down then S/G water levels would decrease to cause reactor to trip on S/G level low-low.

Answer C is incorrect. The feed pumps will increase speed vice slowing down. If the feed pumps did slow down then Low S/G water levels cause a Reactor trip directly. This, in turn, causes a Main turbine trips. The Main Turbine only has a direct S/G Hi-Hi level trip.

Answer D is incorrect. The Feed pumps will increase speed due to the pressure mismatch but the master speed controller will limit speed to prevent an overspeed trip.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
SBK LOP L8046I, Steam Generator Water Level control system.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8062I 07 RO (As available)

Question Source: Bank # 26717
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 11

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	055EA2.02	
	Importance Rating	4.4	4.6

K/A 055 = Station Blackout
EA2 Ability to determine or interpret the following as they apply to a Station Blackout:
EA2.02 RCS core cooling through natural circulation cooling to S/G cooling

Proposed Question:

The following plant conditions exist:

- The plant was operating at 100 % power when a Loss of Off-Site power occurred.
- Ten minutes later, the following conditions exist:
    - o SG A Pressure; 1130 psig and stable.
    - o SG B Pressure; 1125 psig and stable.
    - o SG C Pressure; 1130 psig and stable.
    - o SG D Pressure; 1125 psig and stable.
    - o RCS Pressure; 2250 psig and stable.
    - o T-hot; 580°F in all 4 loops and trending down slowly.
    - o Core Exit TC's; 585°F and trending down slowly.
    - o T-cold; 561°F in all 4 loops and stable

Based on the above indications, what is the status of Natural Circulation?

A. Does exist; heat removal is being maintained by the condenser steam dumps.

B. Does NOT exist; heat removal may be established by opening the condenser steam dumps.

C. Does exist; heat removal is being maintained by atmospheric steam dumps.

D. Does NOT exist; heat removal may be established by opening the atmospheric steam dump valves.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. Natural Circulation is indicated by adequate subcooling, stable SG pressures, RCS Hot leg temperatures slowly decreasing, core exit thermocouples decreasing and RCS cold leg temperatures at saturation for this SG pressure. Students should determine that the secondary heat sink is being provided by the atmospheric steam dumps by SG pressures at approximately at the ASDV lift setpoint (~1125#) and knowledge that the LOP would have interrupted the required condenser steam dumps AC emergency bus power as well as tripped the required circulation water pumps.

Answer A is incorrect. Natural circulation does exist; but the SG side pressure is much higher than that indicative of condenser steam dump operation, as well as the loss of circulation water pumps caused by the LOP.

Answer B is incorrect. Indication of Natural circulation exists, and the condenser steam dumps would not be available when the emergency buses are not available or the circulation water pumps are tripped.

Answer D is incorrect. Indications of Natural Circulation exists. The SG pressures given also indicate that the atmospheric steam dump valves should already be open.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ES-0.1, Reactor Trip Response, Attachment G, Conditions Indicating Natural Circulation

Proposed references to be provided to applicants during examination: None

Learning Objective: L1225I07RO (As available)

Question Source: Bank # 23162
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.3
55.43 43.5

Comments:

Question 12

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	056AA1.31	
	Importance Rating	3.3	3.3

K/A 056 = Loss of Off-site Power
AA1. Ability to operate and / or monitor the following as they apply to the Loss of Offsite Power: AA1.31 PZR heater group control switches

Proposed Question:

The following events have occurred:

- The crew is performing a plant shutdown from 40% Power.
- A manual Safety Injection is actuated due to a Steam Generator Tube Rupture.
- After the Reactor/Turbine Trip a complete Loss of Offsite Power occurs.
- Both E5 and E6 energize from their associated diesel generators and EPS sequencing has completed up to step 9 for both buses.
- PZR level is 18% and slowly Increasing.

Once RMO is reset which of the following describes the mode of operation of the Pressurizer Heaters?

A. Backup heaters groups A & B can be energized in manual ONLY.

B. Control group heaters are energized in auto because of the >5% PZR level deviation.

C. Backup heaters groups A & B are energized in auto because of the >5% PZR level deviation.

D. Control group heaters energize in auto after cycling the Control Switch to OFF and back to NORMAL AFTER START.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. Pressurizer backup heater groups A & B are the only bank of heaters connected to the emergency buses. They are also configured such that when the Emergency Power Sequencer is energized they can only be used in manual.

Answer B is incorrect. The question states that final pressurizer level is 18%, which would be >5% PZR level deviation from program, but the control group heaters are not available when the Emergency Power Sequencer is energized. Additionally the Backup heaters are energized with the 5% deviation.

Answer C is incorrect. The final condition of pressurizer level at 18% does provide a >5% PZR level deviation, but the backup heaters groups A & B will not energize in auto with the Emergency Power Sequencer still activated.

Answer D is incorrect. The Control group heaters are reset after the bus is returned to normal power supply by cycling the Control Switch to OFF and back to NORMAL AFTER START but this will not work at this point.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8022I, Pressurizer and PRT. Pressurizer Heater Logic diagrams 1-NHY-503750, 1-NHY-503749, 1-NHY-503751.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8022I 06RO (As available)

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:
55.41 41.7
55.43 ____

Comments:

Question 13

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	058AK1.01	
	Importance Rating	2.8	3.1

K/A 58 = Loss of DC Power
AK1. Knowledge of the operational implications of the following concepts as they apply to Loss of DC Power: AK1.01 Battery charger equipment and instrumentation

Proposed Question

The following plant conditions exist:

- The plant is operating at 100% power.
- Following a crew brief, Technical Specification action statements have been entered to transfer 125 VDC Bus 11A to its ALTERNATE battery supply in accordance with OS1048.13, "Vital Bus 11A Operation".
- Due to a human performance error, the Nuclear Systems Operator missed several procedure steps and failed to note that battery charger, 1-EDE-BC-1A, was NOT connected to DC Bus 11A.
- The normal battery supply breaker was then opened.

Given these conditions what operational implication is a direct result of these actions?

A. Loss of Turbine Trip Control.

B. Loss of Normal Feedwater Control.

C. Loss of Emergency Diesel Generator Stop Capability.

D. Loss of Pressurizer Power Operated Block Valve Control.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. These conditions are outlined as a caution in OS1048.13 and result in a Loss of Vital Bus 11A. As a result of this bus loss, Feedwater Control is lost and will result in a plant trip.

Answer A is incorrect because this would be a result of non-vital DC bus loss.

Answer C is incorrect because the loss of DC Bus 11A results in a loss of start capability vs. stop capability.

Answer D is incorrect because the Block valve is unaffected, however the PORV control power is lost.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1248.01, Loss of Vital 125VDC Bus, OS 1048.13, Vital Bus 11A Operation

Proposed references to to be provided to applicants during examination: None

Learning Objective: L1189I15RO Describe the effects of losing Vital DC Power on the following equipment. L8020I23RO Describe the importance of 125VDC power for the proper auto operation of the EDG and EPS. (As available)

Question Source: Bank # 26909
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History:
Last NRC Exam: Used on 2005 NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.8 /41.10
55.43 ____

Comments:

Question 14

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	062 G 2.4.11	
	Importance Rating	4.0	4.2

K/A 062 = Loss of Nuclear Service Water
G 2.4 = Emergency Procedures/Plan
G 2.4.11, Knowledge of abnormal condition procedures

Proposed Question:

The following conditions exist:

- The crew has entered OS1216.01, "Degraded Ultimate Heat Sink" due to a loss of all ocean pumps on train "A" Service Water.
- A Tower Actuation Signal (TA) has been actuated for train "A" Service Water.
- At step 6, "Restore SW to Secondary Loads", the crew is directed to place both Ocean SW pumps in PULL TO LOCK and place the Cooling Tower Pump in NA-START.

What is the basis for this step?

A. Allows the operator to reset the Tower Actuation signal.

B. Allows the Cooling Tower Pump discharge valve to reopen on a LOP.

C. Ensures the Cooling Tower Pump breaker will remain closed on a LOP.

D. Prevents a Tower Actuation from occurring if system pressure decreases.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Placing both Ocean pumps in PTL and the Cooling tower pump in Normal After Start/Start generates a TA signal. This signal is required to make the Cooling Tower Pump discharge valve automatically reopens on a LOP.

Answer A is incorrect. The step is designed to ensure a Tower Actuation signal is present.

Answer C is incorrect. The Cooling Tower Pump breaker will open on the LOP, and reclose when sequenced on by the EPS.

Answer D is incorrect. The SW train is already aligned to the Cooling Tower, so another Tower Actuation signal would not be effective.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1216.01, "Degraded Ultimate Heat Sink" step 6. Loops and logics diagrams 1-NHY-503960, 1-NHY-503962, 1-NHY-503979.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8037I 06RO, 12RO and 13RO (As available)

Question Source: Bank # 26883
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 15

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	065AK3.04	
	Importance Rating	3.0	3.2

K/A 065 = Loss of Instrument Air
AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: AK3.04 Cross-over to backup air supplies

Proposed Question:

Which of the following statements is correct concerning the design basis for the backup gas supplies to these components?

A. The ASDVs are provided with a backup supply of bottled Nitrogen that, on loss of normal instrument air supply, is selected via Main Control board switch to allow continued valve operation. Sufficient pressure is provided for RCS cooldown to RHR entry conditions.

B. The Turbine Driven EFW pump steam supply valves are provided with a supply of bottled Nitrogen that supports their dual function as a containment isolation valve and their EFW function. With B/U nitrogen less than the required pressure the valves are INOPERABLE for T.S. 3.6.3, Containment Isolation, as well as T.S. 3.7.1.2, Auxiliary Feedwater.

C. The PCCW temperature control valves are provided with a supply of bottled Nitrogen. This feature allows isolation of the non-seimic portions of the system so that the safety function of the PCCW system is not compromised in the event of a failure in the non-seismic part of the system.

D. The Feed Water Isolation valves have an internal supply of Nitrogen gas. This allows the valves to perform their containment isolation function even in the event of a loss of Normal Instrument Air supply.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The bases for TS 3.7.1.6, Atmospheric relief valves, states "The ARVs are provided with a pressurized gas supply that, on loss of pressure in the normal instrument air supply, automatically supplies nitrogen to operate the ARVs. The nitrogen supply is sized with sufficient pressurized gas to operate the ARVs for the time required for RCS cooldown to RHR Entry conditions. The system design requires the operators to use the "jog switches" on the Main control board when operating under these circumstances. Both IA and Nitrogen bottles are lined up in this mode and a set of check valves are used to "auctioneer" the highest control gas pressure to the valves.

Answer B is incorrect. The bases for TS 3.7.1.2 states that the Turbine Driven EFW pump steam supply valves 1) have a supply of bottled Nitrogen and, 2) are dual function (EFW supply and Containment Isolation) valves. The backup nitrogen only supports their function as a containment isolation valve. With B/U nitrogen less than the required pressure the valves are INOPERABLE for T.S. 3.6.3, Containment Isolation, which would require failing the valves CLOSED, which would then INOP them for T.S. 3.7.1.2, Auxiliary Feedwater.

Answer C is incorrect. The PCCW temperature control valves are provided with a supply of bottled Nitrogen, but this feature allows 10 full strokes of the temperature control valves during a loss of instrument air pressure. The valves that provide automatic isolation of the non-seismic portions of the system on low head tank level are not equipped with backup nitrogen supplies.

Answer D is incorrect. The Feed Water Isolation valves do have an internal supply of Nitrogen gas, but this a "stand-alone" gas volume that is the normal motive force to CLOSE the valves. No backup gas is provided for this function.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Bases for T.S. 3.6.3, 3,7,3, and 3.7.1.6.

Proposed references to be provided to applicants during examination: None

Learning Objective: ____________________ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A, new question.
Exam Bank History N/A, new question.
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis ______

10 CFR Part 55 Content: 55.41 41.5, 41.10
55.43 ______

Comments: AK3. Knowledge of the reasons for the following responses as they apply to the Loss of Instrument Air: AK3.04 Cross-over to backup air supplies

Question 16

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	W/E04EA2.1	
	Importance Rating	3.4	4.3

K/A W/E04 = LOCA outside Containment
EA2. Ability to determine and interpret the following as they apply to the (LOCA Outside Containment): EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question:

The following plant conditions exist:

- A Reactor trip and Safety Injection have occurred.
- The crew transitioned to E-1, "Loss of Reactor or Secondary Coolant" based on Elevated Containment Pressure.
- The crew notes a valid ORANGE path on the Emergency Coolant Recirculation Status Tree and transitions to ECA-1.2, "LOCA Outside Containment".
- After closing RH-V14, "RHR Train 'A' Discharge to RCS", and RH-V22, "RHR Train 'A' Discharge Cross-Connect", the following conditions exist:
  - RCS pressure is 1100 psig and INCREASING
  - ECCS flow is DECREASING

Which of the following describes the expected procedural transition from ECA-1.2?

A. Secure Train "A" CBS and RHR pumps then transition to ES-1.1, "SI Termination" to terminate Safety Injection.

B. Secure Train "A" CBS and RHR pumps then transition to E-1, "Loss of Reactor or Secondary Coolant" for additional recovery actions.

C. Re-Open RH-V-14 and RH-V-22 and continue efforts to isolate the leakage source in ECA-1.2, "LOCA Outside Containment".

D. Close RH-V-26, "RHR Train "B" Discharge to RCS", and RH-V-21, RHR Train "B" Discharge Cross-Connect. Continue efforts to isolate the leakage source in ECA-1.2, "LOCA Outside Containment".

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Initial plant conditions describe a LOCA in containment, followed by a subsequent LOCA outside containment (i.e. RHR pump seal failure). Step 2 of ECA-1.2 isolates the "A" RHR train in an attempt to isolate potential leakage. The effectiveness of the efforts are evaluated using RCS wide range pressure response. Because RCS WR pressure increased after closing RH-V-14 and 22 the crew should secure the pumps associated with train "A" then transition back to E-1 to evaluate plant status.

Answer A is incorrect. The leak appears to be stopped, but the evaluation to transition to ES-1.1 in order to terminate and/or reduce SI flow will be made from E-1, not directly from ECA-1.2.

Answer C is incorrect. The increase in RCS pressure and the corresponding decrease in ECCS flow as pressure increases support that the leak was isolated and no further actions to isolate any leakage outside containment are required at this time. If the candidate incorrectly determined that the leak was not isolated then "A" train RHR would be realigned and additional actions to isolate the leak would continue in ECA-1.2.

Answer D is incorrect. Even if the crew incorrectly determined that the leak did not appear to be isolated "A" RHR would be realigned before isolating the "B" RHR.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ECA-1.2, LOCA Outside Containment, step 3.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1209I 05RO (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A, new question.
Exam Bank History N/A, new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 17

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	W/E11EA2.1	
	Importance Rating	3.4	4.2

K/A W/E11 = Loss of Emergency Coolant Recirculation
EA2. Ability to determine and interpret the following as they apply to the (Loss of Emergency Coolant Recirculation):EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question:

The following plant conditions exist:

- The plant has suffered a LOCA in containment.
- When evaluating recirculation capability in E-1, "Loss of Reactor or Secondary Coolant," the crew discovered there was no electrical power to either train containment sump recirculation valve, CBS-V-8 or CBS-V-14.
- The crew transitioned to ECA-1.1, "Loss of Emergency Coolant Recirculation."
- While checking for Cold Leg Recirculation Conditions in step 1 a NSO reset the breaker for CBS-V-8 and power is now available.
- CBS-V-14 breaker could not be reset and the valve remains CLOSED.
- The RWST volume is 125,000 gallons.

What action is required?

A. Transition to ES-1.3, "Transfer to Cold Leg Recirculation," step 1.

B. Transition to E-1, "Loss of Reactor or Secondary Coolant," procedure and step in effect.

C. Stop the "B" CBS and RHR pumps and remain in ECA-1.1, "Loss of Emergency Coolant Recirculation".

D. Stop the "B" CBS and RHR pumps and transition to E-1, "Loss of Reactor or Secondary Coolant," procedure and step in effect.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The Operator action summary states that if emergency recirculation capability is restored during the performance of this procedure then the recovery actions should continue in the procedure and step in effect. In this case the crew should transition back to E-1, "Loss of Reactor or Secondary Coolant".

Answer A is incorrect. The given RWST level is still above the value that directs a transition to ES-1.3, "Transfer to Cold Leg Recirculation".

Answer C is incorrect. When power was restored to the "A" containment sump isolation valve then the crew would transition back to E-1. The "B" CBS and RHR pumps would only be stopped at this point if the RWST level was below the semi-automatic swapover point.

Answer D is incorrect. Although the crew does transition to E-1, "Loss of Reactor or Secondary Coolant," which is the procedure and step in effect, the "B" CBS and RHR pumps would only be stopped if the RWST level was below the semi-automatic swapover point.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ECA-1.1, Operator Action Summary and Step 8.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1209I 02RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7, 41.10
55.43 43.5

Comments:

Question 18

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	W/E05EA1.2	
	Importance Rating	3.7	4.0

K/A W/E05 = Loss of Secondary Heat Sink
EA1. Ability to operate and / or monitor the following as they apply to the (Loss of Secondary Heat Sink): EA1.2 Operating behavior characteristics of the facility.

Proposed Question:

The following plant conditions exist:

- The Reactor has tripped from 100% Power.
- Both Emergency Feedwater Pumps CANNOT be started.
- The SUFP has tripped due to a motor fault.
- The crew is performing actions in FR-H.1, "Response to Loss of Secondary Heat Sink".

Why is it essential to initiate bleed and feed cooling immediately if conditions are met?

A. Seabrook PORV Flow to Power ratio is such that timely initiation is required to prevent RCS pressurization from lifting the Pressurizer Safety Valves.

B. Seabrook PORV Flow to Power ratio is such that timely initiation is required to ensure bleed and feed cooling will be effective at maintaining core cooling.

C. If the SGs reach dry-out conditions the Steam Generator primary to secondary differential pressure will be exceeded resulting in SG tube sheet failure.

D. If the SGs reach dry-out conditions the Steam Generator Feed Nozzles may fail from the severe thermal stress that would result when feed is finally restored.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Seabrook PORV Flow to Power ratio is 115 lbm/hr/Mwt. Bleed and feed cooling must be initiated before SG dryout in order to ensure adequate core cooling.

Answer A is incorrect. The effectiveness of bleed and feed is tied to timely initiation of the strategy. The RCS heat up and pressurization to Pressurizer safety valve setpoints would occur past the point where bleed and feed can assure core cooling.

Answer C is incorrect. The primary concern of allowing the SGs to reach dry-out conditions would be the RCS heatup and pressurization, which would be followed by RCS mass loss through the PORVs and Pressurizer safeties at their design setpoints.

Answer D is incorrect. Concerns about the thermal stresses when re-introducing feed flow to Steam Generator Feed Nozzles in SGs under dryout conditions become secondary to the benefits of establishing a heat sink. The procedure directs feeding one dry generator at a time in order to limit the damage to one generator at a time, but always directs operators to choose feeding a dry generator above continuing on bleed and feed.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse background document for FR-H.1, Discussion

Proposed references to be provided to applicants during examination: None

Learning Objective: L1412I 12RO, L1211I 03RO (As available)

Question Source: Bank #
Modified Bank # 16358 (Note changes or attach parent)
New

Original TEB 15358

The following plant conditions exist:

- The Reactor has tripped from 100% Power.
- Both Emergency Feedwater Pumps can NOT be started.
- The SUFP has tripped due to a motor fault.
- The crew is performing actions in FR-H.1, "Response to Loss of Secondary Heat Sink".

Why is it essential to initiate bleed and feed cooling immediately if conditions are met?

A. To prevent lifting the Pressurizer safety Valves.

B. To ensure bleed and feed cooling will be effective at maintaining core cooling.

C. To prevent a tube rupture from excessive primary to secondary differential pressure which may result if the SGs boil dry.

D. To prevent a complete dry-out of the Steam Generators and the severe thermal stress that would result when feed is finally restored.

Proposed Answer: B

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Modified from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 19

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	001AK2.01	
	Importance Rating	2.9	3.2

K/A 001 = Continuous Rod Withdrawal
AK2. Knowledge of the interrelations between the Continuous Rod Withdrawal and the following: AK2.01 Rod bank step counters

Proposed Question:

The following plant conditions exist:

- The plant is at 80% Power while performing repairs to a Moisture Separator Reheater.
- The Rod Control system is in AUTO.
- Both Control Bank D Group 1 and group 2 step counters indicate 208 steps.
- All DRPI indicators for Control Bank D indicate 204 steps.
- Control Bank D Rods begin continuously withdrawing.
- The Crew has determined that the Rod withdrawal was not required and placed Rod Control to MANUAL.
- The rod motion stopped.
- The Control Bank D, Group 1 step counter indicates 220 steps.
- The Control Bank D, Group 2 step counter indicates 219 steps.
- DRPI indication for all Control Bank D rods is 216 steps.
- No Rod Control Alarms have been received.

Based on these plant conditions, what can be determined about the Rod Position Indicating Systems?

A. The Rod Position indicating systems are working as designed.

B. Both Group step counters were indicating erroneously before the rod motion.

C. The Group 1 step counter has counted one extra step demand that did not occur.

D. The Control Bank "D" DRPI indication for Group one Control Rods has not stepped to 222 steps as designed.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. Initial group step counters indicated 208 steps for both groups. Initial DRPI indication was 204 steps for both groups. DRPI has a (+/-) 4 step accuracy and the initial height was between DRPI indication of 204 steps and 210 steps. An actual rod height of 208 steps would be within the (-) 4 step tolerance. The final rod height given was group 1 step counter at 220 steps, group 2 step counter at 219 steps and DRPI indication for all Control Bank D rods is 216 steps. Actual rod height would be between the DRPI indications of 222 and 216 steps. Both final group step positions would both be within the (-) 4 step accuracy of DRPI. The Rod Position indicating systems are working as designed.

Answer B is incorrect. If given Group step counter indications were erroneous then this should have resulted in a mismatch with DRPI since the given heights were at the limits of DRPI accuracy.

Answer C is incorrect. The additional step indicated on Group 1 step counter (as compared to group 2) still remains within the DRPI accuracy so there is no indication that an extra step occurred.

Answer D is incorrect. The given final Group one Control Rod height of 220 steps is within the (+/-) tolerance of the given final DRPI indication of 216 steps.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1210.03 Continuous Control Rod Withdrawal, Lesson SBK LOP L8023I "Rod Position Indication".

Proposed references to be provided to applicants during examination: None

Learning Objective: L8032I 04RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 20

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	005 AK3.04	
	Importance Rating	3.4	4.1

005 = Inoperable/Stuck Control Rod
AK 3 Knowledge of the reasons for the following responses as they apply to the Inoperable/Stuck Control Rod.
AK3.04 Tech-Spec Limits for inoperable rods.

Proposed Question:

The following plant conditions exist:

- The plant is at 70% power with a power increase in progress.
- Rod H8 in Control Bank D is indicating below the other Control Bank D rods.
- Control Bank D group one and group two step counters are at 140 steps.
- Rod H8 is indicating 126 steps by DRPI.
- The other rods in Control Bank D are indicating 138 steps by DRPI.
- Rod H8 does not move when the other Bank D rods move.
- I&C investigation reveals that ALL fuses for rod H8 are blown.

What Technical Specification action is required in response to these conditions?

A. Shutdown Margin must be verified within 1 hour and be in HOT STANDBY within 6 hours.

B. The power increase may continue provided the Rod Insertion limits are maintained and the remainder of the rods in the bank with the inoperable rod are aligned to within +/- 12 steps of the inoperable rod.

C. The Power level must stay below 75% and the remainder of the rods in the bank with the inoperable rod must be aligned to within +/- 12 steps of the inoperable rod. The High Flux Neutron trip setpoint must be lowered to 85% within the next 4 hours.

D. The power increase may continue provided that the Rod Insertion limits are maintained and within 1 hour the remainder of the rods in the bank with the inoperable rod are aligned to within +/- 12 steps of the inoperable rod. The Inoperable rod must be restored to OPERABLE within 72 hours.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The students should recognize that a rod which does not drop to the bottom of the core when all it's fuses are blown is stuck. Action "A" of TS 3.1.3.1 requires that Shutdown Margin must be verified within 1 hour and the plant placed in HOT STANDBY within 6 hours with one or more rods inoperable due to excessive friction or mechanical interference.

Answer B is incorrect. This distractor restates TS 3.1.3.1 action b, 2. for a single rod that is trippable but inoperable and the remaining rods in the bank can be aligned within 12 steps while staying above the RIL.

Answer C is incorrect. This distractor restates TS 3.1.3.1 action b, 3. for a single rod that is trippable but inoperable AND at a low enough height that action b, 2 can not be complied with.

Answer D is incorrect. This distractor restates TS 3.1.3.1 action c. for more than one rod that is trippable but inoperable.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Tech Spec 3.1.3.1, Movable Control Assemblies.

Proposed references to be provided to applicants during examination: None

Learning Objective: ______ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A, new question
Exam Bank History N/A, new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 21

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	051AA2.02	
	Importance Rating	3.9	4.1

AA2. Ability to determine and interpret the following as they apply to the Loss of Condenser Vacuum: AA2.02 Conditions requiring reactor and/or turbine trip

Proposed Question

The following plant conditions currently exist:

- The plant was at 100% power.
- Condenser Vacuum is 22.5 in. hg and slowly decreasing.
- Turbine load reduction is in progress.
- Turbine load is 360 MWE.

Which of the following actions should be taken by the crew?

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

B. Continue the load reduction to increase condenser vacuum to > 25 in. hg.

C. Immediately trip the reactor and go to E-0, REACTOR TRIP OR SAFETY INJECTION.

D. Continue the load reduction and if vacuum remains > 22.4 in. hg. remove the turbine generator from service per OS1000.06, POWER DECREASE.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. Load is 360 MWE which is the maximum load reduction allowed by ON1233.01. Procedure directs manual RX trip if vacuum continues to degrade when load had been decreased to this point.

Answer A is incorrect. The Loss of Condenser Vacuum abnormal procedure calls for a manual reactor trip, not a manual turbine trip. Answer is plausible as the megawatt load in the stem is below the P-9 setpoint, when the turbine could be tripped without a reactor trip.

Answer B is incorrect. The conditions in the question stem call for a reactor trip.

Answer D is incorrect. A load decrease below 360 MWE can damage the main turbine so it should not be conducted. A reactor trip is required.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ON1233.01, Loss of Condenser Vacuum.

Proposed references to be provided to applicants during examination: None

Learning Objective: Lesson L1188I, Objective L1188I08 (if available)

Question Source:	Bank #	29963	
	Modified Bank #	______	(Note changes or attach parent)
	New	______	

Question History:
Last NRC Exam: Question from 2007 NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content:
55.41 41.10
55.43 43.5, 45.13

Comments:

Question 22

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	060AK2.02	
	Importance Rating	2.7	3.1

K/A 060 = Accidental Gaseous Radwaste Release
AK2. Knowledge of the interrelationships between the Accidental Gaseous Radwaste Release and the following: AK2.02 Auxiliary building ventilation system

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- The Control Board Monitor notices a slow increase in the WRGM effluent radiation level.

Which of the following conditions could be the cause for this?

A. PAH-FN-42A, PAB Aux Supply Fan, has started on area high temperature and a small leak has developed at the Letdown Radiation Monitor.

B. EAH-FN-180B, Charging Pump Room Return Air Fan, has started on low differential pressure to the ECCS Vaults.

C. WAH-FN-13B, Waste Process Building Exhaust Fan, is in service and a leak has developed in the Steam Space of the running Primary Drain Tank Degassifier.

D. PAH-DP-1003, PAB/FSB Balance damper, has been left in the FUEL HANDLING position after the Fuel Handling Building Ventilation was returned to the NORMAL mode.

Proposed Answer: C

Explanation (Optional):

Answer C is correct because WAH flows to the plant stack and by the WRGM before it is discharged.

Answer A is incorrect because PAH-FN-42 exhausts directly outside the building and will not flow past the WRGM.

Answer B is incorrect because total air flow out of the plant exhaust would not change with the start of EAH-FN-180A or 180B.

Answer D is incorrect because changing position of PAH-DP-1003 would change the amount of air flowing to the FSB but would not change the overall amount of air discharging out of the plant vent.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8040I, Primary Building Heating and Ventilation Systems.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8040I 03RO, L8040I 04RO, ____________ (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5, 41.10
55.43 ____

Comments:

Question 23

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	W/E14EK2.1	
	Importance Rating	3.4	3.7

K/A W/E14 = High Containment Pressure
EK2. Knowledge of the interrelations between the (High Containment Pressure) and the following: EK2.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features

Proposed Question:

The following plant conditions exist:

- A LOCA has occurred.
- Containment pressure is 21 psig and decreasing.
- Safety Injection has just been RESET per step 10 of E-1, Loss of Reactor or Secondary Coolant.
- The white light associated with the operation of CBS-V8, Containment Recirculation Sump Isolation valve, is ON (MCB AF bench section).

What is the significance of this light?

A. Power is available, enabling CBS-V8 to automatically open when the RWST reaches LO-LO level.

B. A "P" signal is present, enabling CBS-V8 to automatically open when the RWST reaches LO-LO level.

C. An "S" signal is present, enabling CBS-V8 to automatically open when the RWST reaches LO-LO level.

D. Either RC-V22 or 23 is closed, enabling CBS-V8 to automatically open when the RWST reaches LO-LO level.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The "S" signal is retained within the CBS sump semi-automatic swapover logic in order make the swapover occur, but still allow operators to reset the "S" signal to re-align plant equipment as required after the initial automatic "SI" alignment is complete. The retained "S" signal indicated by the white light will enable CBS-V8 to automatically open when the RWST reaches LO-LO level.

Answer A is incorrect. The white light does not provide any indication of electrical power availability to CBS-V8.

Answer B is incorrect. The white light indicated the retained "SI" signal, not the presence of a "P" signal is present. The "P" signal is not required for the semi-automatic swapover to occur.

Answer D is incorrect. The loop suction valves, RC-V22 or 23, must be closed to MANUALLY open CBS-V8, but the automatic opening on RWST LO-LO level coincident with the retained "SI" signal will occur regardless of their position.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8035I. CBS-V-8 (14) schematic 1-NHY-310900 B84a.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8035I13RO (As available)

Question Source: Bank # 22051
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 24

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	076 G 2.2.25	
	Importance Rating	3.2	4.2

K/A 076 = High Reactor Coolant Activity
2.2 Equipment Control
2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question:

Technical Specification 3.4.8. "Reactor Coolant System – Specific Activity" requires the unit to be placed in Hot Standby with Tavg less than 500°F if the specific activity of the Reactor Coolant exceeds the gross radioactivity limits.

What is the basis for the limitation on the Reactor Coolant System Specific Activity?

A. Ensures that Steam Generator tube integrity is maintained in the event of a main steam line rupture.

B. Limit is based on Operating Experience with SG tube degradation mechanisms that result in tube leakage.

C. Prevents the release of activity should a steam generator tube rupture, since the saturation pressure of the reactor coolant is below the lift pressure of the Steam Generator Code Safety Valves.

D. Ensures that the resultant offsite dose will be limited to a small fraction of the 10CFR50.67 guidelines in the event of a Steam Generator tube rupture with the assumed 1 gpm primary to secondary leakage.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The bases for TS 3/4.4.8 states the limit ensures that the resultant 2-hour dose at the site boundary will not exceed an appropriately small fraction of the 10CFR50.67 guidelines in the event of a Steam Generator tube rupture with the assumed 1 gpm primary to secondary leakage.

Answer A is incorrect. The bases for TS 3/4.6.2, Operational Leakage, discusses maintaining SG tube integrity by limiting SG leakage limit to 1 gpm for each SG and 500 gpd for all SGs in total. This is done to ensure that the resultant radioactive release from any event resulting in a steam discharge to the atmosphere is less than the safety analysis limits. If the RCS specific activity specification is not met then the bases for the postulated amount of release in a main steam line break is no longer valid, but limiting the RCS activity is NOT tied to ensuring SG tube integrity.

Answer B is incorrect. The bases for TS 3/4.6.2, Operational Leakage identifies the limit of 150 gpd as based on operating experience associated with SG tube degradation mechanisms that result in tube leakage.

Answer C is incorrect. The bases for TS 3/4.4.8 cites this distractor as the reason why RCS temperature is reduced to less than 500 degrees, not as the bases for the activity limit.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
TS bases for 3/4 .4.8, Reactor Coolant Specific Activity and bases for 3/4.6.2, Operational Leakage.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1181I07RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5 /41.7
55.43 43.2

Comments:

Question 25

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	W/E02EK1.2	
	Importance Rating	3.4	3.9

K/A W/E02 = SI Termination
EK1. Knowledge of the operational implications of the following concepts as they apply to the (SI Termination): EK1.2 Normal, abnormal and emergency operating procedures associated with (SI Termination)

Proposed Question:

The following plant conditions exist:

- The plant has experienced a small break LOCA.
- Total EFW flow has been throttled to 550 GPM based on RCS temperature less than 557 degrees.
- The Crew has transitioned from E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant" and now to ES-1.1, "SI Termination" in order to reduce ECCS flow.
- Plant parameters are as follows:
  - Containment pressure is 1.5 psig and slowly decreasing.
  - Pressurizer level is 40% and increasing.
  - RCS Subcooling is 43° and stable.
  - RCS pressure is 1950 psig and stable.
- The crew is terminating SI.
- After placing the first CCP in standby, RCS pressure starts to slowly decrease.

Which of the following describes the Crew response to these conditions?

A. Restart the CCP and go to E-0, "Reactor Trip or Safety Injection".

B. Transition to ES-1.2, "Post LOCA Cooldown and Depressurization".

C. Restore normal charging path and control charging flow to maintain Pressurizer Level.

D. Initiate Safety Injection and transition to E-1, "Loss of Reactor or Secondary Coolant".

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Step 2 of ES-1.1 directs a transition to ES-1.2, "Post LOCA Cooldown and Depressurization", if RCS pressure is not stable or increasing after securing the first Centrifugal Charging Pump.

Answer A is incorrect. The OAS page of ES-1.1 only directs manual restart of ECCS pumps if RCS subcooling is lost or Pressurizer level can not be maintained greater than 7%. In this case, because the RCS PRESSURE is SLOWLY decreasing a transition to ES-1.2 is more appropriate. The OAS also would direct a transition back to E-1, step 1 if RCS inventory was truly challenged, not E-O, "Reactor Trip or Safety Injection".

Answer C is incorrect. Restoring normal charging path and controlling charging flow to maintain Pressurizer Level is not directed in either ES-1.1 or ES-1.2.

Answer D is incorrect. In this case a SLOW degradation of RCS pressure after stopping the first Centrifugal Charging pump would only require a transition to ES-1.2. Additionally, if the RCS pressure did lead to re-initiation criteria the procedure directs "Manually Start ECCS pumps as Required" not "Initiate Safety Injection".

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ES-1.1 "SI TERMINATION", step 2 and OAS page.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1226I 05 RO (As available)

Question Source: Bank # 6082
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.8 /41.10
55.43 ______

Comments:

Question 26

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	W/E13EA1.1	
	Importance Rating	3.1	3.3

W/E13 = Steam Generator Overpressure
EA1 : Ability to operate and/or monitor the following as they apply to the (Steam Generator Overpressure):
EA1.1 Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Proposed Question:

The disabling switch at CP-450A (in the "A" Emergency Diesel Room) for MS-PV-3001, "A SG ASDV Train A Power", is placed in DISABLE. How does this action affect the control scheme for this valve?

A. The valve can be operated ONLY locally in manual override at the valve itself.

B. The valve can be operated from the Train "A" Remote Safe Shutdown panel (CP-108A).

C. The valve can be operated ONLY from the Train "B" Remote Safe Shutdown panel (CP-108B).

D. The valve can be operated from the Main Control board "Jog Switch" (OPEN/CLOSE/POSITION MAINTAIN Switch).

Proposed Answer: D

Explanation (Optional):

Answer D is correct: The ASDVs are dual train valves, but only ASDV "A" and "C" have switches at the "A" RSS panel and only ASDV "B" and "D" have switches at the "B" RSS panel. All four valves have non-safety related control power and safety related control power from BOTH trains available in the control room. The disabling switches are designed to remove the non-safety control system and the affected train safety valve controls from the circuit when taken to disable. The Train "A" disabling switch will remove Train "A" safety related control power and the non-safety related control power. After this the valve can be operated from the Main Control board "Jog Switch" (OPEN/CLOSE/POSITION MAINTAIN Switch) using the safety related control power from the other ("B") train.

Answer A is not correct. In addition to local manual override at the valve itself, the valve can be operated as described in Answer D.

Answer B is incorrect. The safety related control power for Train "A" Remote Safe Shutdown panel (CP-108A) was disconnected when going to "disable".

Answer C is incorrect. There is no control switch for the "A" ASDV at the Train "B" Remote Safe Shutdown panel (CP-108B).

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8041I. Schematic 1-NHY-310841 sheets E2T/8a and E2U/15

Proposed references to be provided to applicants during examination: None

Learning Objective: L1211I 05RO (As available)

Question Source: Bank # 25973
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5 /41.10
55.43 ______

Comments:

Question 27

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	W/E08EK1.3	
	Importance Rating	3.5	4.0

K/A W/E08 = Pressurized Thermal Shock
EK1. Knowledge of the operational implications of the following concepts as they apply to the (Pressurized Thermal Shock): EK1.3 Annunciators and conditions indicating signals, and remedial actions associated with the (Pressurized Thermal Shock).

Proposed Question:

The following plant conditions exist:

- A Steam break inside containment occurred at 100% power.
- All systems operate as designed.
- Intact SG NR level is approximately 15%.
- EFW flow has been throttled to 100 gpm to intact SGs.
- The crew is in FR-P.1, Response to Imminent Pressurized Thermal Shock Conditions.
- RCS temperature is stable.
- RCS pressure is stable.
- The pressurizer heater control group is energized.
- The crew has commenced a temperature soak.

What action can be performed by the crew within the next hour?

A. Place auxiliary spray in service.

B. Increase RCS subcooling from 45 degrees to 100 degrees.

C. Increase EFW flow to 300 gpm per SG to raise NR levels to 50%.

D. Place RHR in shutdown cooling mode and commence cooldown to Mode 5.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. Step 23 of FR-P.1 verifies adequate RCS depressurization. If normal Pressurizer spray is not available the RNO directs placing auxiliary spray in service.

Answer B is incorrect. FR-P.1 attempts to maintain a minimum amount of subcooling, something less than 50 degrees. In order to increase RCS subcooling the operators would have to either raise RCS pressure or cooldown the RCS, both of which would be undesirable in this situation.

Answer C is incorrect. FR-P.1 throttles EFW flow to help control the cooldown. Any increase in EFW flow at this point will restart the cooldown before the completion of the RCS temperature soak.

Answer D is incorrect. If RCS pressure and temperature were low enough it would be permissible to place RHR in shutdown cooling mode, but a cooldown to Mode 5 would not be allowed until the RCS temperature soak was completed.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
FR-P.1, Response to Imminent Pressurized Thermal Shock Conditions.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1208I 05RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.8 /41.10
55.43 ____

Comments:

Question 28

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	003K1.04	
	Importance Rating	2.6	2.9

K/A 003 = Reactor Coolant Pump System
K1 Knowledge of the physical connections and/or cause-effect relationships between the RCPS and the following systems: K1.04 CVCS
Idea: 21101

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- Letdown is in service at 100 gpm.
- The #1 Seal Leakoff flow for each Reactor Coolant Pump is 3 gpm
- The Seal Injection flow for each Reactor Coolant Pump is 8 gpm.

What is the Charging flow, as indicated on the Main control board, required to maintain a constant pressurizer level?

A. 100 gpm.

B. 112 gpm.

C. 120 gpm.

D. 132 gpm.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Indicated Charging flow should equal indicated letdown flow plus seal return flow for a stable mass balance in the RCS. In this case: 112gpm = 100gpm + 4(3gpm)

Answer A is incorrect. This distractor assumed that indicated charging flow would equal indicated letdown flow.

Answer C is incorrect. This distractor assumed that charging flow equaled letdown flow plus seal injection flow minus seal leakoff flow. 120gpm = 100gpm + 4(8gpm) – 4(3gpm)

Answer D is incorrect. This distractor assumed that charging flow equaled letdown flow plus seal injection flow. 132 gpm = 100gpm + 4(8gpm)

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1001.05, Reactor Coolant Pump Operation, Lesson SBK LOP L8024I, Chemical Volume and Control System

Proposed references to be provided to applicants during examination: None

Learning Objective: L8024I 09RO (As available)

Question Source: Bank # 21101
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.2 to 41.9/ 45.7 to 45.8
55.43 ______

Comments:

Question 29

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	003K5.04	
Importance Rating	3.2	3.5

K/A 003 = Reactor Coolant Pump System
K5 Knowledge of the operational implications of the following concepts as they apply to the RCPS: K5.04 Effects of RCP shutdown on secondary parameters, such as steam pressure, steam flow, and feed flow

Proposed Question:

The following plant conditions exist:

- The plant is operating at 40% power.
- A problem with Reactor Coolant Pump "B" requires that it be shutdown.
- All Steam Generator Narrow Range levels are at 50%.
- All Steam Generator steam flows are equal.

What is the expected response upon stopping the "B" RCP?

A. "B" SG NR level will swell and steam flow will increase.

B. "B" SG NR level will shrink and steam flow will decrease.

C. "A", "C" and "D" SG NR level and steam flow will remain unchanged.

D. "A", "C" and "D" SG NR level will shrink and steam flow will decrease.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. When the RCP is secured flow will first stop, then reverse in that loop but at a much lower mass flow rate. This resulting low flow will significantly decrease the steam flow from the "B" SG. As the steam flow decreases from the affected Steam Generator it's steam pressure increases. The pressure increase decreases the size of the steam bubbles in the riser section of the steam generator, allowing more volume to move from the downcomer into the SG tube area. This is seen as a decrease in SG (downcomer) level.

Answer A is incorrect. The affected SG level will shrink on a downpower, not swell.

Answer C is incorrect. The remaining Steam Generators will increase steam flow, so their level will swell.

Answer D is incorrect. The remaining Steam Generators will increase steam flow, so their level will swell.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L1181I, Reactor Coolant Pump Malfunction.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1181I05RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.6
55.43 ____

Comments:

Question 30

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	004A3.02	
	Importance Rating	3.6	3.6

K/A 004 = Chemical Volume and Control System
A3 Ability to monitor automatic operation of the CVCS, including: A3.02 Letdown isolation

Proposed Question:

Given the following plant conditions:

- The plant is at 100% power.
- An inadvertent Phase "A" Containment Isolation occurs on train "A".
- The Reactor has NOT tripped and Safety Injection (SI) has NOT actuated.

Based on these conditions, how does this isolation affect the Chemical and Volume Control System (CVCS) Letdown path?

A. CS-V-149, "Letdown Heat Exchanger IRC Isolation", and CS-V-145, "Letdown Regen Heat Exchanger Isolation", have closed.

B. RC-LCV-459, "Letdown Regen Heat Exchanger Isolation", and CS-V-145, "Letdown Regen Heat Exchanger Isolation", have closed.

C. CS-V-149, "Letdown Heat Exchanger IRC Isolation", CS-V150, "Letdown Heat Exchanger ORC Isolation", and CS-V-145, "Letdown Regen Heat Exchanger Isolation", have closed.

D. RC-LCV-459, "Letdown Regen Heat Exchanger Isolation", RC-LCV-460, "Letdown Regen Heat Exchanger Isolation" and CS-V-145, "Letdown Regen Heat Exchanger Isolation", have closed.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. CS-V-149 is the only valve in the letdown path that will receive a Train "A" phase "A" isolation signal, but CS-V-145 is interlocked to CS-V-149 so that V-145 will close if V-149 closes to prevent lifting the letdown line relief valve.

Answer B is incorrect. RC-LCV-459 is interlocked to close CS-V-145 if LCV-459 closes, but LCV-459 only automatically closes on low Pressurizer level, not a Phase "A" signal or if V-145 closes.

Answer C is incorrect. CS-V150 would only close if a Train "B" phase "A" signal were also received.

Answer D is incorrect. Both RC-LCV-459 and RC-LCV-460 are interlocked to CS-V-145 such that if either LCV-459 or LCV-460 go close then V-145 will close, but RC-LCV-459 or LCV-460 would only automatically close on a low Pressurizer level, not closure of V-145.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Schematic drawings 1-NHY-310891, sheets B7/a and E97/11b, 1-NHY-310882, sheets E89/17a and E89/1b. Lesson Plan SBK LOP L8024I, Chemical Volume and Control System.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8024I 10RO (As available)

Question Source: Bank #
Modified Bank # 26935 (Note changes or attach parent)
New

Original 26935

Given the following plant conditions:

- The plant is at 100% power.
- An inadvertent Phase "A" Containment Isolation occurs on both trains.
- The Reactor has NOT tripped and Safety Injection (SI) has NOT actuated.

Based on these conditions, how does this isolation affect the Chemical and Volume Control System (CVCS) Letdown path?

A. Only letdown isolation valves (CS-V-149, CS-V-150) and CS-V-145 have closed.
B. Only Letdown isolation valves (RC-LCV-459, RC-LCV-460) and CS-V-145 have closed.
C. The Letdown isolation valves (CS-V-149, CS-V150) and CS-V-145 and Reactor coolant Pump seal return valves (CS-V-167 and CS-V-168) have closed.

D. The Letdown isolation valves (CS-V-149, CS-V150) and CS-V-145 and Reactor coolant Pump seal return valves (CS-V-167 and CS-V-168), and the containment charging isolation valves (CS-V-142 and CS-V-143) have closed.

Proposed Answer: C

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Modified from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 31

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	005K4.11	
	Importance Rating	3.5	3.9

K/A 005 = Residual Heat Removal
K4 Knowledge of RHRS design feature(s) and/or interlock(s) which provide for the following: K4.11 Lineup for low head recirculation mode (external and internal)

Proposed Question:

The following conditions exist:

- A large break LOCA has occurred.
- The Crew is in E-1, "Loss of Reactor or Secondary Coolant".
- SI has just been reset.
- VAS alarm "RWST LO-LO" has annunciated.
- RWST level is 120,000 gals.

Which of the following describes the response of the valve re-alignment for the CBS pump suction flowpath?

A. CBS-V-8 and CBS-V-14, “Containment Sump Isolation Valves”, will not OPEN automatically because the SI signal is reset and CBS-V-2 and CBS-V-5, “RWST Suction Isolation valves”, cannot be manually CLOSED.

B. CBS-V-8 and CBS-V-14, “Containment Sump Isolation Valves”, must now be manually OPENED because the SI signal is reset and CBS-V-2 and CBS-V-5, “RWST Suction Isolation valves”, automatically CLOSE .

C. CBS-V-8 and CBS-V-14, “Containment Sump Isolation Valves”, will OPEN automatically because the SI signal is still present in the CBS suction swapover circuit. CBS-V-2 and CBS-V-5, “RWST Suction Isolation valves”, are manually CLOSED when CB-V-8 and CBS-V-14 are FULL OPEN.

D. CBS-V-8 and CBS-V-14, “Containment Sump Isolation Valves”, will OPEN automatically because the SI signal is still present in the CBS suction swapover circuit. CBS-V-2 and CBS-V-5, “RWST Suction Isolation valves”, will automatically CLOSE when CB-V-8 and CBS-V-14 are FULL OPEN.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The containment swapover logic retains an SI signal which is reset separately from the "master" SI signal. The Containment Sump Isolation Valves OPEN automatically with this retained SI signal coincident with RWST level at the LO, LO level setpoint of 120,478 gallons. The swapover is "semi-automatic so the RWST Suction Isolation valves are manually CLOSED after the Containment Sump Isolation Valves are open.

Answer A is incorrect. It is the "master" SI signal that is reset in E-1 so the Containment Sump Isolation Valves should still OPEN automatically. The RWST Suction Isolation valves can not be manually OPENED unless the Containment Sump Isolation valves are closed, but there is no interlock which prevents closing the valves.

Answer B is incorrect. The Containment Sump Isolation Valves should automatically open. There is no automatic repositioning of the RWST Suction Isolation valves.

Answer D is incorrect. The Containment Sump Isolation Valves will OPEN automatically because the SI signal is still present in the CBS suction swapover circuit, but the RWST Suction Isolation valves must be manually closed after the Containment Sump Isolation Valves go FULL OPEN.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Schematic 1-NHY-310900 sheet B52b, 1-NHY-310900 sheet B84a.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8035I 13RO (As available)

Question Source: Bank # ____
Modified Bank # 26716 (Note changes or attach parent)
New ____

The following conditions exist:

- A large break LOCA has occurred.
- The Crew is in E-1, "Loss of Reactor or Secondary Coolant".
- SI has just been reset.
- VAS alarm "RWST LO-LO" has annunciated.
- RWST level is 120,000 gals.

Which of the following describes the response of the CBS pump containment recirculation sump valves?

A. RWST level has reached semi-automatic swapover setpoint. Valves will not open because the SI signal has been reset.

B. RWST level has reached semi-automatic swapover setpoint. Valves will open because an SI signal is still present to the CBS suction swapover circuit.

C. RWST level has not reached semi-automatic swapover setpoint. Valves should be manually aligned per ES-1.3, "Cold Leg Recirculation".

D. RWST level has not reached semi-automatic swapover setpoint. Crew should wait for level to decrease below the setpoint.

Answer B

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.7
	55.43	

Comments:

Question 32

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006K3.03	
	Importance Rating	4.2	4.4

K/A 006 = Emergency Core Cooling
K3 Knowledge of the effect that a loss or malfunction of the ECCS will have on the following: K3.03 Containment

Proposed Question:

The following plant conditions exist:

- The plant has suffered a large break LOCA in containment
- Both "A" and "B" RHR pumps have failed.
- All other ECCS pumps are running as designed.
- The crew is preparing to transition from E-1, "Loss of Reactor or Secondary Coolant" to ECA-1.1, "Loss of Emergency Coolant Recirculation"
- The crew has also identified a valid ORANGE path on the Containment Status tree which directs an entry to FR-Z.1, "Response to High Containment Pressure".

Which of the following describes the actions the crew will take to address Containment Pressure conditions?

A. The crew will use ECA-1.1 to STOP all CBS pumps regardless of Containment Pressure.

B. The crew will use FR-Z.1 to START or STOP the number of CBS pumps running based on RWST level.

C. The crew will use ECA-1.1 to START or STOP the number of CBS pumps running based on Containment Pressure.

D. The crew will use FR-Z.1 to START all available CBS pumps until the ORANGE Containment status tree is cleared.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. An ORANGE path on the containment status tree would become the higher priority according to the rules of usage, however there is a caution prior to step 2 of FR-Z.1 that directs the crew to run CBS pumps in accordance with the direction in ECA-1.1. In ECA-1.1 the number of CBS pumps running is changed based on Containment Pressure.

Answer A is incorrect. ECA-1.1 provides the overriding guidance in this case but it does not stop the CBS pumps. The number of running CBS pumps is limited to preserve as much available RWST inventory for later use by the ECCS injection pumps, but RWST inventory is not for ECCS pumps a higher priority.

Answer B is incorrect. The number of running CBS pumps is dictated by ECA-1.1 in this case. If FR-Z.1 did provide guidance it becomes more plausible to assume that procedure would vary the number of CBS pumps running based on RWST level.

Answer D is incorrect. FR-Z.1 does START all available CBS pumps until the ORANGE Containment status tree is cleared, but the students need to recognize that ECA-1.1 has a higher priority as stated in the caution prior to step 2.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Caution prior to step 2 in FR-Z.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1212I 08 RO (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New: New

Question History:
Last NRC Exam: N/A new question
Exam Bank History: N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 33

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	006A4.11	
	Importance Rating	4.2	4.3

K/A 006 = Emergency Core Cooling
A4 Ability to manually operate and/or monitor in the control room: A4.11 Overpressure protection system

Proposed Question:

The following plant conditions exist:

- The crew is performing a plant shutdown and cooldown for a refueling outage.
- The plant is in Mode 5.
- RCS temperature is 195 degrees and RCS pressure is 325 PSIG.
- Wide Range Pressure transmitter RC-PT-403 is removed from service for calibration.
- Train "A" RHR is aligned to the shutdown cooling mode.
- RC-V-88, "Train 'B' RHR Outboard Loop Isolation Valve", has been CLOSED for emergent Train "B" RHR pump seal repair work.

The Control Board Monitor is determining if RCS overpressure protection is provided on the Shift Technical Specification logs. What is the status of required RCS overpressure protection in this alignment?

A. The required Overpressure protection is met by verifying both PORV block valves are OPEN with both PORVs OPERABLE in the LTOP mode.

B. The required Overpressure protection is met by verifying that both PORV block valves are OPEN with the "A" PORV OPERABLE in the LTOP mode, and the "B" PORV OPERABLE in MANUAL.

C. The required Overpressure protection is met by verifying the train "B" PORV block valve is OPEN with the "B" PORV OPERABLE in the LTOP mode, and both RC-V-22, "Train 'A' RHR Inboard Loop Isolation Valve" and RC-V-23, "Train 'A' RHR Outboard Loop Isolation Valve" are OPEN.

D. The required Overpressure protection is met by verifying the train "A" PORV block valve is OPEN with the "A" PORV OPERABLE in the LTOP mode, and both RC-V-22, "Train 'A' RHR Inboard Loop Isolation Valve" and RC-V-23, ""Train 'A' RHR Outboard Loop Isolation Valve" are OPEN.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. TS 3.4.9.3 requires at least two overpressure protection devices; PORVs in LTOP mode, or RHR suction reliefs. The "A" PORV block valve OPEN with the "A" PORV OPERABLE in the LTOP mode would provide one path. Both RC-V-22, "Train 'A' RHR Inboard Loop Isolation Valve" and RC-V-23, ""Train 'A' RHR Outboard Loop Isolation Valve" OPEN would provide a second flowpath to the "A" RHR suction relief.

Answer A is incorrect. Removing RC-PT-403 from service would disable the LTOP capability of the "B" PORV making it inoperable. Only the "A" PORV would be available as Operable Overpressure protection.

Answer B is incorrect. "A" PORV can be OPERABLE in MANUAL in modes 1, 2, and 3 (TS 3.4.4) but not for LTOP purposes.

Answer C is incorrect. The "B" PORV would not be OPERABLE in the LTOP mode with PT-403 removed from service. The only OPERABLE pressure protection would be the "A" train RHR suction relief with both RC-V-22 and RC-V-23 OPEN.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
T.S. 3.4.9.3, Overpressure Protection Systems.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8022I 13RO (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43

Comments:

Question 34

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	007 G 2.4.20	
	Importance Rating	3.8	4.3

K/A 007 = Pressurizer Relief Tank/Quench tank system
2.4 Emergency Procedures / Plan
2.4.20 Knowledge of the operational implications of EOP warnings, cautions, and notes.

Proposed Question:

The following plant conditions exist:

- The Crew has initiated a Safety Injection in response to a Steam Generator tube rupture.
- A Loss of Off-site 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 states: "*The PRT may rupture if a PZR PORV is used to depressurize the RCS. This may result in abnormal containment conditions.*"

Which of the following statements describes the implications of this caution?

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.

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.

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.

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 Without Pressurizer Pressure Control.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. While the PRT rupture disk 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 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 degraded such that a transition to another procedure was required.

Answer A is incorrect. It is true that the PRT rupture disk may fail before RCS pressure is reduced to ruptured SG pressure and that this may result in increasing Containment radiation and humidity. The crew would only transition if RCS subcooling or Pressurizer level can not be maintained, and the transition would be to ECA-3.1.

Answer C is incorrect. The caution prior to step 18 does warn that cycling of the PZR PORV should be minimized to avoid failure of the PRT rupture disc, but the background document states that this it to minimize the chance of failure of a PORV. A transition to ECA-3.3, SGTR Without Pressurizer Pressure Control would be required if the crew determined that a PORV and Auxiliary spray was not available (because the PRT rupture disc was going to fail).

Answer D is incorrect. The RNO for step 18 would only use Auxiliary Spray if 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, SGTR Without Pressurizer Pressure Control if no PORV or Auxiliary spray was available.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Background document for E-3, for cautions prior to step 18 (step number in Seabrook document, W background document step 17).

Proposed references to be provided to applicants during examination: None

Learning Objective: L8022I 11RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 41.10	
	55.43 43.5	

Comments:

Question 35

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008K1.02	
	Importance Rating	3.3	3.4

K/A 008 = Component Cooling Water System
K1 Knowledge of the physical connections and/or cause-effect relationships between the CCWS and the following systems: K1.02 Loads cooled by CCWS

Proposed Question:

The following plant conditions exist:

- RCS temperature is 320°F on Train “A” RHR cooling.
- “C” RCP is operating.
- It is noted that the “A” PCCW head tank level is decreasing more rapidly than normal.

Which heat exchanger has a tube leak?

A. “A” RHR HX.

B. CVCS Regenerative HX.

C. “C” RCP Thermal Barrier HX.

D. CVCS Seal Water Return HX.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The student should recognize that RCS pressure should be less that 350 psig to place RHR in service, but greater than 325 psig to keep a RCP in service. For this pressure band the RCS side of the CVCS Seal Water Return heat exchanger would be approximately equal to VCT pressure of 25# and the only choice at a pressure lower than normal PCCW pressure of 100 psig.

Answer A is incorrect. The process side of the RHR heat exchanger would be at RCS pressure plus RHR pump discharge pressure. Any leakage in this heat exchanger would cause PCCW head tank level to increase.

Answer B is incorrect. The CVCS Regenerative heat exchanger is cooled by charging system flow.

Answer C is incorrect. The RCP Thermal Barrier heat exchanger (in the pump seal cavity) should be pressurized to RCS pressure and is a separate fluid system from the PCCW system.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L80336I, Primary Component Cooling Water System. OS1212.01, PCCW System Malfunction, Attachment B; Heat Loads

Proposed references to be provided to applicants during examination: None

Learning Objective: L8036I 04 RO (As available)

Question Source: Bank # 6093
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.2 to 41.9
55.43 ______

Comments:

Question 36

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	008A2.02	
	Importance Rating	3.2	3.5

K/A 008 = Component Cooling Water System
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CCWS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 High/low surge tank level

Proposed Question:

What conditions are required to allow filling of the PCCW system from the Fire Protection system per OS1212.01, "PCCW Malfunction"?

A. A fill from the Fire Protection header is permitted only if SW-P-374, "Service Water Booster pump", is not available.

B. Emergency fill from the Fire Protection system must only be used when the affected PCCW train is the only remaining loop available to cool safety related components.

C. Fire Protection fill is ONLY allowed for short duration operation because prolonged use of chlorinated makeup water will lead to chloride corrosion cracking in the PCCW piping.

D. A Fire Protection fill may be performed to restore cooling to at least one Charging Pump oil cooler if BOTH the running Charging pump and the standby Charging pump have lost PCCW cooling.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. There is a caution at the beginning of attachment D, "Emergency Fill from Fire Protection System", that states this must only be used when affected PCCW system is the only remaining loop available to cool safety related components.

Answer A is incorrect. The service water booster pump is used to fill the fire protection system if the normal fire protection system supply is lost. The normal fire protection system is the preferred source.

Answer C is incorrect. The caution at the beginning of attachment D, "Emergency Fill from Fire Protection System", also states that prolonged use of chlorinated makeup water source may lead to chloride stress corrosion cracking in PCCW system but this is not the limiting consideration. The fill path from Fire Protection would only be used for one loop and only if it was the only loop available to cool Safety Related components. If the PCCW loop was required then the fill (and FP water) would be used as long as necessary to support the required heat sink.

Answer D is incorrect. The use of Fire Protection when BOTH the Running Charging pump and the Standby Charging pump have lost cooling concerns use of the fire protection system as a cooling heat sink for the individual Charging pump lube oil coolers, not for filling an entire PCCW loop with fire protection water.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1201.01, RCP Malfunction, OS1212.01, PCCW Malfunction.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8036I 12RO (As available)

Question Source: Bank # ___
Modified Bank # ___ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 37

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	010A2.01	
	Importance Rating	3.3	3.6

K/A 010 = Pressurizer Pressure Control
A2 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:A2.01 Heater failures

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power
- Alarm "D4327, PZR PRESSURE LOW & BU HTRS ON" is received
- The PSO notes that the Master Pressure Controller output is 43%

Which of the following is a correct response to this situation in accordance with station procedures?

A. The Master Pressure Controller output is failing LOW which will DEENERGIZE the Control Group heaters and the Backup heaters. Operator action is required to ENERGIZE Control Group and Backup heaters as required to maintain RCS pressure.

B. The Master Pressure Controller output is failing HIGH which has caused the Pressurizer spray valves to modulate OPEN. Operator action will be required to MANUALLY DECREASE Master Pressure controller output to return RCS pressure to program.

C. A Pressurizer Spray valve has failed CLOSED and the Master Pressure Controller output has INCREASED to ENERGIZE Backup heaters. Operator action will be required to MANUALLY INCREASE Master Pressure controller output to return RCS pressure to program.

D. The Control Group heaters have failed and the Master Pressure Controller output has DECREASED to ENERGIZE Backup heaters. The Backup Heaters will cycle ON and OFF to control RCS pressure. Operator action will be required to stabilize RCS pressure by placing Backup heaters ON to force Pressurizer Spray.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The VAS alarm D4327 actuates based on master pressure controller output and only indicates a demand for the backup heaters to energize, not that they have actually energized. In the case of the Control Group heaters failing the Master Pressure Controller output would lower as RCS pressure lowered until the Backup heaters energized. The Backup Heaters will cycle ON and OFF between 43% output to 45% output to control RCS pressure. This will continue unless the heaters are taken to ON, which would raise RCS pressure control until the master pressure controller throttled open Pressurizer Spray valves.

Answer A is incorrect. The VAS alarm D4327 actuates based on master pressure controller output and only indicates a demand for the backup heaters to energize, not that they have actually energized. If the Master Pressure Controller output actually was failing low it would energize the Control Group heaters and the Backup heaters vice deenergize them.

Answer B is incorrect. The students should recognize that a Master Pressure Controller output of 43% is below the normal output of 52% so the controller is not failing HIGH. If the controller was failing high it would cause the Pressurizer spray valves to modulate OPEN and Operator action would be required to stop the RCS pressure decrease.

Answer C is incorrect. If pressurizer sprays had been "forced" then a Pressurizer Spray valve failing closed might cause the Master Pressure Controller to energize Backup heaters but the output would decrease not increase.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8022I, Pressurizer and PRT Operation.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8027I 08RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	___
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 41.5	
	55.43 43.5	

Comments:

Question 38

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	012K6.02	
	Importance Rating	2.9	3.1

K/A 012 = Reactor Protection system
K6 Knowledge of the effect of a loss or malfunction of the following will have on the RPS:
K6.02 Redundant channels

Proposed Question:

The following plant conditions exist:

- PZR pressure channel 455 has failed LOW.
- All associated bistables have been placed in trip.
- PZR pressure channels 457/456 have been selected for control and backup respectively
- Now PZR channel 457 fails LOW.

Which of the following describes the effect, if any, this failure has on PORV operation in the present mode?

A. Only PORV 456A is prevented from opening automatically.

B. Only PORV 456B is prevented from opening automatically.

C. Both PORVs are prevented from opening automatically.

D. Both PORVs will open automatically when required.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The failure low of channel 457 will prevent operation of the "A" PORV because it has been selected as the actuating channel (457/456 pushbutton), and prevent operation of the "B" PORV because it is always interlocked to the "B" PORV as the "arming" channel.

Answer A is incorrect. The "B" PORV will also not operate because PT-457 always provides the arming for the "B" PORV.

Answer B is incorrect. PT-457 became the "actuating" channel for the "A" PORV when it was selected by the 457/456 pushbutton.

Answer D is incorrect. Neither PORV will open automatically when required.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1201.06, Pressurizer Pressure Instrument/ Component Failure , Caution prior to step 3.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1182I 14RO (As available)

Question Source: Bank # 22676
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 39

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	013K6.01	
	Importance Rating	2.7	3.1

K/A 013 = Engineering Safety Features Actuation System
K6 Knowledge of the effect of a loss or malfunction on the following will have on the ESFAS: K6.01 Sensors and detectors

Proposed Question:

The following plant conditions exist:

- Plant is at 100% power.
- Containment Pressure Channel III, SI-PT-936, failed HIGH.
- The Crew tripped or bypassed the channel bistables as required to comply with Technical Specifications for this failure.
- 10 hours later Containment Pressure Channel II, SI-PT-935, subsequently failed HIGH.

How will the plant respond as a result of the PT-935 failure?

A. No Automatic Actuations will occur.

B. Containment Building Spray will automatically actuate.

C. Main Steamline Isolation and Safety Injection will automatically actuate.

D. Containment Building Spray, Main Steamline Isolation, and Safety Injection will automatically actuate.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. When a containment pressure channel fails Technical Specification 3.3.2, table 3.3-3, item 2c, action 15 requires Hi-3, Containment Spray actuation, placed in bypass within 6 hrs. This now means that 2 of 3 channels need to actuate in order to initiate containment spray (vice 1 of the remaining 3 to make up 2 of 4 logic). Since only one more channel failed the CBS actuation only has 1 of 3 made up. The Hi-1 and Hi-2 channels for the failed instrument are taken to trip. This means 1 of 3 was already made up, and a second channel failing will make up the 2 of 3 logic. Main Steamline Isolation and Safety Injection will automatically actuate.

Answer A is incorrect. CBS will not actuate, but MSI and SI will actuate.

Answer B is incorrect. CBS will NOT automatically actuate.

Answer D is incorrect. CBS will NOT actuate, but MSI and SI will automatically actuate.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
TS 3.3.2 Engineered Safety Features Actuation Instrumentation, Table 3.3-3, items 1.c, 2.c, and 4.c. Westinghouse functional diagram 1-NHY-509048.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8035I (As available)

Question Source: Bank #
Modified Bank # 18870 (Note changes or attach parent)
New

Original TEB 18870

The following plant conditions exist:

- Containment Pressure Channel SI-PT-937 failed high during the previous shift.
- The crew performed the ACTIONS required by Technical Specifications.
- Containment Pressure channel SI-PT-935 subsequently fails high.

How will the plant respond as a result of the PT-935 failure?

A. No Automatic actuations occur as a result of the failure.

B. Containment building Spray will automatically actuate.

C. Main Steam Line Isolation and Safety Injection will automatically actuate.

D. Containment Building Spray, Main Steam Line Isolation and Safety Injection will automatically actuate.

Answer: A

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.7
	55.43	____

Comments:

Question 40

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	022A3.01	
	Importance Rating	4.1	4.3

K/A 022 = Containment Cooling
A3 Ability to monitor automatic operation of the CCS (containment cooling system), including: A3.01 Initiation of safeguards mode of operation

Proposed Question:

The following sequence of events occur:

- Plant trip, Safety Injection and Loss of Offsite Power occurs simultaneously.
- 15 minutes later, Safety Injection is reset.
- 20 minutes later, Remote Manual Override is reset.
- All equipment operates as designed.

What is the status of the previously running Containment Structure Cooling fans?

A. The fans are off and must be manually restarted.
B. The fans restarted when Safety Injection was reset.
C. The fans were restarted by the Emergency Power Sequencer at step 3.
D. The fans are off and can only be restarted when Offsite Power is restored.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The fans initially trip on the loss of PCCW flow caused by the Loss of Offsite power. The PCCW pumps are sequenced back on by the EPS early so that flow is available by step 3. In this case the fans would not start at step 3 because they are also locked out by a SI. They can be MANUALLY restarted when both SI and RMO are reset, but there is no automatic restart of the CAH coolers.

Answer B is incorrect. The fans are locked out with an SI coincident with a LOP. They can be MANUALLY restarted when both SI and RMO are reset, but there is no automatic restart of the CAH coolers.

Answer C is incorrect. In this case the fans would still be locked out by the SI signal which had not been reset.

Answer D is incorrect. The fans can be MANUALLY restarted when both SI and RMO are reset, but there is no automatic restart of the CAH coolers.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8038I, Containment Air Handling.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8038I 04 RO (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 41

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	026 G 2.4.4	
	Importance Rating	4.5	4.7

K/A 026 = Containment Spray System
2.4 Emergency Procedures / Plan
2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry level conditions for emergency and abnormal operating procedures
Proposed Question

Which condition requires a RED path entry to FR-Z.1, "Response to High Containment Pressure"?

A. Containment pressure is 19 PSIG and no CBS pump is running.

B. Containment pressure is 18 psig and Containment Building Sump level is 2 feet.

C. Containment pressure is 50 psig with all required ESF equipment functioning as designed.

D. Containment pressure is 20 psig and RMW-V-30, "Makeup Water to Containment", is not CLOSED.

Proposed Answer D

Explanation (Optional):

Answer D is correct. If Containment pressure is greater than 18 psig and any phase A or phase B penetration is not isolated then this meets a FR-Z.1 RED condition. The RMW-V-30, "Makeup Water to Containment", penetration only has an "A" train valve so this condition would make up a FR-Z.1 RED CSF.

Answer A is incorrect. The Containment integrity status tree does not use CBS pumps running or not running in the decision flowpath.

Answer B is incorrect. Containment Building Sump level is only evaluated for ORANGE path conditions in the Containment integrity status tree.

Answer C is incorrect. All ESF equipment functioning as designed would signify that all phase "A" and phase "B" penetrations are isolated. As long as this is true and containment pressure is less than 52 psig then it is not possible to reach a RED path on the Containment integrity status tree.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse Functional evaluation status tree F-0.5, Containment (Z)

Proposed references to be provided to applicants during examination: None

Learning Objective: L1212I16RO (As available)

Question Source: Bank #
Modified Bank # 16070 (Note changes or attach parent)
New

Original TEB 16070
Which condition requires a RED path entry to FR-Z.1, "Response to High Containment Pressure"?

A. Containment pressure is 18 PSIG and no CBS pump is running.

B. Containment pressure is 4.3 psig and rising at a rate of 1 psig per minute.

C. Containment pressure is 50 psig with all required ESF equipment functioning as designed.

D. Containment pressure is 20 psig and the letdown line containment penetration is NOT isolated.

Proposed Answer D

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Modified from bank

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

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

10 CFR Part 55 Content: 55.41
55.43

Comments:

Question 42

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	039A1.06	
	Importance Rating	3.0	3.1

K/A 039 = Main and Reheat Steam System
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the MRSS controls including: A1.06 Main steam pressure

Proposed Question:

The following plant conditions exist:

- The plant is in Mode 3
- The crew is performing a plant cooldown in accordance with OS1000.15, "Refueling Outage Cooldown".
- Tave is 540°F and slowly decreasing.
- PZR pressure is 1850 psig and slowly decreasing.
- P-11 has energized and all blocking actions have been taken.
- S/G pressures are all approximately 950 psig and slowly decreasing.

While attempting to increase the RCS cooldown rate, the BOP operator inadvertently throttles too far open on the Condenser Steam Dumps. All of the "SG B PRESSURE RATE-HI MSI" lights are illuminated, then extinguish.

What automatic function(s), if any, occur as a result of this condition?

A. No automatic actions occur.
B. Safety Injection actuates ONLY.
C. Main Steam Line Isolation actuates ONLY.
D. Main Steam Line Isolation AND Safety Injection Actuation occur.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. After decreasing RCS pressure P-11 the Pressurizer (Low Pressure) Safety Injection and the Steamline (Low Pressure) Safety Injection are blocked. Blocking the Steamline Safety Injection also changes the Main Steam Line (Low Pressure) Isolation from actuating on low pressure to actuating on high rate of change. When the SG B HI RATE MSI lights illuminated this indicated that the rate setpoint had been exceeded. A Main Steam Line Isolation on high rate actuates ONLY.

Answer A is incorrect. A Main Steam Line isolation on hi rate occurred.

Answer B is incorrect. A Main Steam Line Isolation occurred and the Safety Injection Actuation was blocked.

Answer D is incorrect. A Main Steam Line Isolation occurred and the Safety Injection Actuation was blocked.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1000.15, Refueling Outage Cooldown, step 4.2.8, 1-NHY-509046, 1-NHY-509047 sheet 2, 1-NHY-509048

Proposed references to be provided to applicants during examination: None

Learning Objective: L1171I 02 RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments:

Question 43

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	059K3.03	
	Importance Rating	3.5	3.7

K/A 059 = Main Feedwater System
K3 Knowledge of the effect that a loss or malfunction of the MFW will have on the following: K3.03 S/GS

Proposed Question:

Given the following plant conditions:

- The reactor is operating at full power near end of life.
- A Large Feedwater Line break downstream of the Feed Line check valves inside containment occurs.

Which of the following indications would initially distinguish the Feedwater Line break from a Main Steam Line break inside containment?

A. RCS Tave prior to the reactor trip.

B. Containment pressure after the reactor trip.

C. Steam generator pressure after the reactor trip.

D. Steam Generator narrow range level prior to the reactor trip.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. For a large Feedwater Line break Steam Generator narrow range level would sharply decrease prior to the reactor trip. For a Main Steam Line break SG narrow range level would be relatively unaffected, aside from some amount of SG swell for large breaks.

Answer A is incorrect. RCS Tave should initially slowly decrease for both accidents until significant SG level was lost.

Answer B is incorrect. Containment pressure should sharply increase as energy from the SG is released into containment after the reactor trip. The rate of pressure increase may start faster with a steam line break, but both in both accidents the entire SG should blow down into containment so the end amount of energy released should be relatively close.

Answer C is incorrect. In both cases the Steam generator would depressurize to containment pressure after the reactor trip.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse background document for E-2, discussion section.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1412I 05 RO (As available)

Question Source: Bank # 22598
Modified Bank # ____ (Note changes or attach parent)
New ____

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 44

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061A1.04	
	Importance Rating	3.9	3.9

K/A 061 = Auxiliary/Emergency Feedwater System
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the AFW controls including: A1.04 AFW source tank level

Proposed Question:

Which of the following describes how CST level affects post trip operating conditions?

A. Do not operate the SUFP with CST levels less than 215,000 gallons while aligned to the Upper Tap. Level less than 215,000 gallons will cause the SUFP to trip.

B. If CST makeup is unavailable and CST level cannot be maintained greater than 290,000 gallons then Stop ALL RCPs and immediately start a plant cooldown to mode 5.

C. If CST makeup capability is unavailable and CST level is less than 330,000 gallons then immediately start a plant cooldown to place RHR in service in order to ensure plant protection.

D. Use of the Condensate Spill valve is minimized to prevent contamination of the CST. If CST level decreases to below 212,000 gallons, then use of the spill valve is permitted to restore level to 290,000 gallons.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The caution prior to step 4.1.1 in OS1000.11, Post Trip to Hot Standby, states that level less than 215,000 gallons will cause the SUFP to trip.

Answer B is incorrect. ALL RCPs are not stopped in OS1000.11, Post Trip to Hot Standby, because that would cause a loss of Pressurizer Spray flow. If CST makeup is unavailable and CST level cannot be maintained greater than 290,000 gallons then ALL BUT ONE RCP should be secured (OS1000.11, step 4.3.1).

Answer C is incorrect. The CST size is limited at Seabrook, but precaution 3.4 in OS1000.11, Post Trip to Hot Standby, identifies 290,000 gallons (vice 330,000 gallons) as the "line in the sand" below which prompt action should be taken.

Answer D is incorrect. The use of the Condensate Spill valve is minimized as stated, but the level is varied between 250,000 gallons and 330,000 gallons (note prior to 4.1.11, OS1000.11, Post Trip to Hot Standby).

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1000.11, Post Trip to Hot Standby.

Proposed references to be provided to applicants during examination:
None

Learning Objective: L1169I 03RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5
55.43 ____

Comments:

Question 45

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	061A2.05	
	Importance Rating	3.1	3.4

K/A 061 = Auxiliary/Emergency Feedwater System
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the AFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.05 Automatic control malfunction

Proposed Question:

The following plant conditions exist:

- A Reactor trip without SI occurred from 100% power.
- FW-P-37A, The Turbine Driven EFW Pump, is not running.
- MS-V129 indicates OPEN, but the other steam supply valves are closed.
- The EOP directs the crew to "start EFW pump(s) and align valves".

What action should the crew take to start the TDEFW pump?

A. Place the switches for MS-V-393, MS-V-394 and MS-V-395 in the OPEN position.

B. Place the switches for MS-V-393 and MS-V-394 in the OPEN position. Verify that the switch for MS-V-395 is in AUTO.

C. Place the switch for MS-V-395 in the OPEN position. Verify that the switches for MS-V-393 and MS-V-394 are in AUTO.

D. Verify the switches for MS-V-393, MS-V-394 and MS-V-395 are in the AUTO position. IF a valid EFW start demand is received THEN verify all valves OPEN.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. When the switches for MS-V393 and MS-V394 are placed in the OPEN position and the switch for MS-V-395 is in AUTO then MS-V-395 should open automatically after the 28 second time delay. This will allow the steam lines time to properly blow down any accumulated moisture.

Answer A is incorrect. Placing the switches for MS-V393, MS-V394 in OPEN will cause them to open. Placing the switch for MS-V395 in the OPEN position will cause V-395 to open, BUT before the 28 second blowdown is complete. The pump is likely to overspeed as water and steam are sent down the supply pipe.

Answer C is incorrect. Placing the switch for MS-V395 in the OPEN position would open the valve, but MS-V393 and MS-V394 would not open without some other system demand.

Answer D is incorrect. The EOP intended that the pump would be started and EFW flow made available, regardless of the presence of an EFW start signal. Monitoring the system for a valid demand signal does not meet this intent.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Logic Diagram 1-NHY-503674, 1-NHY-503584.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8045I 02 RO and L8045I 04 RO (As available)

Question Source: Bank # 3941
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 46

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062K1.02	
	Importance Rating	4.1	4.4

K/A 062 = AC Electrical Distribution
K1 Knowledge of the physical connections and/or cause-effect relationships between the AC distribution system and the following systems: K1.02 ED/G

Proposed Question:

The following plant conditions exist:

- The plant is in mode 3
- 345 KV bus 2 is out of service for Switchyard design modifications.
- Station electrical loads are supplied by the UATs.
- 345 kV breaker 163 is out for maintenance, with its associated manual disconnects on both sides open.
- All other plant conditions are normal, and no other equipment is out of service.

345 KV breaker 11 is inadvertently opened with the conditions described above (There is no fault on any bus). Which of the following describes the resultant 4.16KV Bus power supply configuration?

A. The electrical lineup did not change, Buses 3, 4, 5 & 6 are still receiving power from the UATs.

B. Buses 3, 4, 5 & 6 are receiving power from the RATs when the busses automatically transferred to their alternate supply.

C. Buses 3 & 4 remain de-energized; buses 5 & 6 de-energize, the EPSs immediately actuate, and buses 5 & 6 are energized by the EDGs.

D. Opening breaker 11 initiates a breaker failure; therefore, there is no power to the UATs or RATs. Buses 3, 4, 5 & 6 de-energize, bus 6 must be re-energized from SEPS.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. Candidates have to recognize that having 345KV bus 2 out of service means that the Reserve Auxiliary Transformers (RATs) were not available. When breaker 163 and breaker 11 both were opened there was also no power to the Unit Auxiliary Transformers (UATs). Buses 3 & 4 would de-energize and remain de-energized. Buses 5 & 6 would initially de-energize, but the Emergency Power Sequencers (EPSs) would immediately actuate (no time delay with no voltage sensed on the RATs), and buses 5 & 6 become re-energized by the Emergency Diesel Generators (EDGs).

Answer A is incorrect. Switchyard breaker 11 and 163 are the breakers that isolate the UATs and the Generator Step Up transformers (GSUs).

Answer B is incorrect. No power is available to the RATs with 345 KV bus 2 unavailable.

Answer D is incorrect. Power is restored to buses 5 and 6 by their respective EDGs. SEPS is not required.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lessons SBK LOP L8011I, 345KV Electrical distribution, SBK LOP L8013I, 4.16 KV Electrical distribution, and SBK LOP L8020I, Emergency Diesel Generator Electrical.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8020I 08 RO and L8013I 14RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.2 to 41.9
55.43 ____

Comments:

Question 47

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	062A3.04	
	Importance Rating	2.7	2.9

K/A 062 = AC Electrical distribution
A3 Ability to monitor automatic operation of the ac distribution system, including: A3.04 Operation of inverter (e.g., precharging synchronizing light, static transfer)

Proposed Question:

During his routine rounds in the 'A' train essential switch gear room the secondary NSO notices that the "Reverse Transfer" lamp is lit on static transfer switch EDE-CP-1E.

What is the condition of EDE-PP-1E?

A. EDE-PP-1E is de-energized.

B. The maintenance supply breaker at EDE-PP-1E has tripped open.

C. The transfer switch has swapped EDE-PP-1E to its alternate supply.

D. The transfer switch has swapped EDE-PP-1E back to its inverter supply.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The static transfer switches for Inverters 1E and 1F constantly monitor the normal and maintenance power supplies, and will shift to the best source (normal seeking). The "reverse transfer" means that the static transfer switch has swapped EDE-PP-1E to its maintenance supply and it will not automatically swap back, even if the normal power supply becomes 'better".

Answer A is incorrect. The static transfer switch has shifted EDE-PP-1E to it's maintenance supply. It is not de-energized.

Answer B is incorrect. The maintenance supply breaker at EDE-PP-1E is now in service.

Answer D is incorrect. A "reverse transfer" means that the Static transfer switch can NOT go back its inverter supply.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8018I, 120 VAC Distribution System.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8018I 09 RO (As available)

Question Source: Bank # 24393
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 48

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	063K2.01	
Importance Rating	2.9	3.1

K/A 063 = DC Electrical Distribution
K2 Knowledge of bus power supplies to the following: K2.01 Major DC loads

Proposed Question:

What, if any, is the effect of a loss of Vital 125 VDC bus 11A on DG-1A?

A. Loss of ALL engine protective tripping capability.

B. Loss of Normal and Emergency engine start circuits.

C. No effect, DG-1A remains FUNCTIONAL from bus 11C.

D. Loss of ALL engine operating parameter indications from any location.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. DG-1A Normal and Emergency engine start circuits are only powered from VDC bus 11A.

Answer A is incorrect. The Diesel can still trip on the mechanical Overspeed trip.

Answer C is incorrect. While bus 11C is also on an "A" train VDC electrical supply, the DG-1A needs bus 11A to be FUNCTIONAL or OPERABLE.

Answer D is incorrect. All local "front standard" indications are still available after a loss of bus 11A.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8020I, Emergency Diesel Electrical, SBK-LOP-L8017I, 125 VDC Distribution System. Abnormal Operating procedure OS1248.01, Loss of a Vital 125 VDC Bus.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1189I02RO (As available)

Question Source: Bank # 30094
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 49

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	064K3.02	
Importance Rating	4.2	4.4

K/A 064 = Emergency Diesel Generators
K3 Knowledge of the effect that a loss or malfunction of the ED/G system will have on the following: K3.02 ESFAS controlled or actuated systems

Proposed Question:

The following sequence of events has occurred:

- A Loss of Offsite Power (LOP) and Safety Injection have actuated.
- The "A" EPS actuated and functioned normally.
- The "B" EPS started the "B" EDG and closed the output breaker but stopped at step 0 of the sequence.

What impact does this have on the operation of "B" train SI/LOP loads?

A. Loads will automatically start after depressing the RMO reset pushbutton.

B. Loads will automatically start after the RMO bypass switch is placed in "BYPASS".

C. Loads must be placed in PTL so they can be manually started after the sequencer is deenergized.

D. Loads must be placed in PTL so they can be manually started after placing the RMO bypass switch in "BYPASS".

Proposed Answer: C

Explanation (Optional):

Answer C is correct. When the emergency power sequencer is de-energized any load with a "standing" auto start signal to them will energize. Procedure ECA-0.0, Loss of All AC Power provides direction in the event that the emergency power sequencer "hangs up" before the sequence is complete. In the RNO for step 5.f the operators are directed to perform step 6 to place all loads in PTL then de-activate the sequencer. Loads are then manually started using the procedure.

Answer A is incorrect. RMO cannot be reset until the sequencer has completed it's sequence.

Answer B is incorrect. The RMO bypass switch is used to reclose a UAT or RAT breaker, and is not related to starting loads on the emergency bus. The loads also would not automatically load without the emergency power sequencer.

Answer D is incorrect. The RMO bypass switch is used to reclose a UAT or RAT breaker, and is not related to manually starting loads on the emergency bus.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ECA-0.0, step 5 and Attachment G.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8020I 21RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 50

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	064A4.09	
	Importance Rating	3.2	3.3

K/A 064 = Emergency Diesel Generators

A4 Ability to manually operate and/or monitor in the control room: A4.09 Establishing power from the ring bus (to relieve ED/G)

Proposed Question:

The following plant conditions exist:

- The plant experienced a loss of off-site power followed 10 minutes later by a safety injection actuation.
- Off-site power was subsequently regained and the operators are in the process of transferring buses E5 and E6 from the Emergency Diesel Generators (EDGs) to an off-site source.

Which of the following describes how the EDGs Governors are placed in the Synchronous ("Speed Droop") mode of operation?

A. Automatically selected when a safety injection signal is present.

B. When the syncroscope selector switch is placed to the UAT position

C. When the Emergency Diesel Generator governor control is adjusted from the MCB.

D. Automatically selected when a Loss of Off-site power occurs.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The emergency Diesel generator control systems place the EDGs in "Speed Droop" when the syncroscope selector switch is placed to the UAT position.

Answer A is incorrect. During an LOP/SI the EDGs are in Isochronous governor control.

Answer C is incorrect. The Emergency Diesel Generator governor controls can be adjusted from the MCB in Isochronous or Speed Droop mode of operation.

Answer D is incorrect. The Emergency Diesel Generator only shifts to Speed Droop when it is synchronized with offsite power.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8020I, Emergency Diesel Generator Electrical.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8020I 05RO (As available)

Question Source: Bank # 20796
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 51

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	073K4.01	
	Importance Rating	4.0	4.3

K/A 073 = Process Radiation Monitoring System
K4 Knowledge of PRM system design feature(s) and/or interlock(s) which provide for the following: K4.01 Release termination when radiation exceeds setpoint

Proposed Question:

Which radiation monitor(s) in HIGH alarm will cause WG-FV-1602, "The Waste Gas Discharge Valve", to close?

A. RM-6503 (1GM631), Inlet Waste Gas Compressor ONLY.

B. RM-6504 (1GM632), Outlet Waste Gas Compressor ONLY.

C. RM-6503 (1GM631), Inlet Waste Gas Compressor, OR RM6504 (1GM632), Outlet Waste Gas Compressor.

D. Both RM-6503 (1GM631), Inlet Waste Gas Compressor, AND RM-6504 (1GM632), Outlet Waste Gas Compressor

Proposed Answer: B

Explanation (Optional):

Answer B is correct. WG-V-1602 is interlocked to RM-6504 (1GM632), Outlet Waste Gas Compressor ONLY.

Answer A is incorrect. If RM-6504 is inoperable then RM-6503 (1GM631), Inlet Waste Gas Compressor can be monitored in lieu of WG system grab samples, but it cannot be used as a substitute for RM-6504 because it does not provide auto isolation.

Answer C is incorrect. Only RM6504 (1GM632), Outlet Waste Gas Compressor, provides isolation.

Answer D is incorrect. Only RM6504 (1GM632), Outlet Waste Gas Compressor, provides isolation.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Logic Diagram 1-NHY-504034. Offsite Dose Control Manual, chapter 5, 5.2, Radiological Gaseous Effluent Monitoring Instrumentation, page A.5-16.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1187I 02 RO (As available)

Question Source: Bank # 22031
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 52

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	076 G 2.1.30	
	Importance Rating	4.4	4.0

K/A 076 = Service Water System
2.1 Conduct of Operations
2.1.30 Ability to locate and operate components, including local controls.

Proposed Question:

The following plant conditions exist:

- The crew has evacuated the control room to the Remote Safe Shutdown panels.
- The crew notes no train "A" Service water pumps running.

What are the required actions to restore "A" train SW flow?

A. Start SW-P-41A from the Remote Safe Shutdown panel. Manually OPEN SW-V-2, "SW PUMP A DISCH ISOLATION", from the Remote Safe Shutdown panel.

B. Start SW-P-41A from it's Bus 5 breaker cubicle. Manually OPEN SW-V-2, "SW PUMP A DISCH ISOLATION" from the Remote Safe Shutdown panel.

C. Start SW-P-41A from the Remote Safe Shutdown panel. SW-V-2, "SW PUMP A DISCH ISOLATION" will automatically OPEN.

D. Start SW-P-41A from it's Bus 5 breaker cubicle. SW-V-2, "SW PUMP A DISCH ISOLATION" will automatically OPEN.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The remote breaker control (used for RSS operation) for SW-P-41A is at the Bus 5 breaker cubicle. No local (RSS) control is required for SW-V-2, "SW PUMP A DISCH ISOLATION" because it will automatically OPEN when the pump is started.

Answer A is incorrect. SW-P-41A can not be started from the Remote Safe Shutdown panel. There is no control for SW-V-2, "SW PUMP A DISCH ISOLATION", at the Remote Safe Shutdown panel.

Answer B is incorrect. SW-P-41A is started from it's Bus 5 breaker cubicle, but SW-V-2, "SW PUMP A DISCH ISOLATION" automatically opens when the pumps starts and there is no Remote Safe Shutdown panel controls for the valve.

Answer C is incorrect. SW-P-41A can not be started from the Remote Safe Shutdown panel, but it is true that SW-V-2, "SW PUMP A DISCH ISOLATION" will automatically OPEN.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Schematic 1-NHY-301107, Sheets AQ3b and CR6a.

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________ (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 53

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078K2.02	
	Importance Rating	3.3	3.5

K/A 078 = Instrument Air System
K2 Knowledge of bus power supplies to the following: K2.02 Emergency air compressor

Proposed Question:

The following plant conditions exist:

- 1-SA-C-137A is running in LEAD, 1-SA-C-137B and 1-SA-C-137C are in LAG.
- No other standby compressor is available
- A loss of Off-site Power has occurred
- The "A" Emergency Diesel repowered it's respective bus
- The "B" Emergency Diesel has failed to start

Which of the following describes the response of the Service Air System following the Loss of Power?

A. 1-SA-C-137A will start when the electrical power is restored. No other compressor will start.

B. 1-SA-C-137C will start when the electrical power is restored. No other compressor will start.

C. Both 1-SA-C-137A and 1-SA-C-137C will start when the electrical power is restored.

D. 1-SA-C-137A will start when the electrical power is restored. 1-SA-C-137C will start if Service air system pressure drops to the LAG setpoint.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. 1-SA-C-137A is powered by US-52 and 1-SA-C-137B is powered from US-63, both with safety related breakers. 1-SA-C-137C is powered from US-21. When the "A" EDG energized it's bus all the train "A" safety related unit substations would be energized (including US-52) so power would be available to 1-SA-C-137A. The compressor features a power outage "auto start" function that starts the compressor when power is lost and subsequently restored.

Answer B is incorrect. 1-SA-C-137C is not powered from an emergency bus.

Answer C is incorrect. Only 1-SA-C-137A will be powered so 1-SA-C-137C will not start. Any compressor that does start will do so as soon as power is restored.

Answer D is incorrect. Only 1-SA-C-137A will be powered and 1-SA-C-137C will not start. 1-SA-C-137A will start as soon as power is restored. The compressors also retain their previous setpoints during a loss of power so, if 1-SA-C-137C did start, it would immediately start (on the power outage "auto start" feature) but only load if Service air system pressure dropped to the LAG setpoint.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8023I, Station Compressed Air Systems.

Proposed references to be provided to applicants during examination: None

Learning Objective: ______________________ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New New

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 54

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	078K4.02	
	Importance Rating	3.2	3.5

K/A 078 = Instrument Air Systems
K4 Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: K4.02 Cross-over to other air systems

Proposed Question:

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

A. AUTOMATICALLY CLOSE at 90 psig decreasing, AUTOMATICALLY OPEN above 93 psig INCREASING.

B. AUTOMATICALLY CLOSE at 80 psig decreasing, AUTOMATICALLY OPEN above 83 psig INCREASING.

C. AUTOMATICALLY CLOSE at 90 psig decreasing, resets to allow MANUAL OPENING above 93 psig INCREASING.

D. AUTOMATICALLY CLOSE at 80 psig decreasing, resets to allow MANUAL OPENING above 83 psig INCREASING.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. SA-V-92 will close when pressure is less than 90 psig. The isolation signal resets when pressure is greater than 93 psig. The reset allows manual opening via control switch as specified in ON1242.01.

Answer A is incorrect because SA-V-92 & 93 do not reopen automatically when pressure is regained.

Answers B & D are both incorrect because the listed set-points are incorrect. Additionally, B is incorrect because the valves do not automatically reopen.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ON1242.01 Loss of IA, L8023I Compressed Air Lesson

Proposed references to be provided to applicants during examination: None

Learning Objective: L8023I16RO Describe the impact on plant operation if both IA loops within either compressed air system should become depressurized (As available)

Question Source: Bank # 23157
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Question used on 2005 NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 55

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	103A4.04	
	Importance Rating	3.5	3.5

K/A 103 = Containment System
A4 Ability to manually operate and/or monitor in the control room: A4 04 Phase A and phase B resets

Proposed Question:

The following Plant conditions exist:

- A Steam Generator has faulted in containment.
- Containment pressure reached 20 psig and has now dropped to 8 psig.
- The Crew has transitioned to ES-1.1, SI TERMINATION because the criteria to reduce ECCS flow was met.
- The "SI" signal is reset.
- The Crew is attempting to reset the "T signal" and "P signal" as directed by procedure.

How does the containment pressure transient affect the ability to reset these signals?

A. Both the "T signal" and the "P signal" CANNOT be reset at the current Containment Pressure.

B. Both the "T signal" and the "P signal" CAN be reset at any time in this transient regardless of Containment pressure.

C. The "T signal CAN be reset at the current Containment Pressure. The "P signal" CANNOT be reset at the current containment pressure.

D. The "T signal" CANNOT be reset at the current Containment Pressure. The "P signal" CAN be reset at the current containment pressure.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The resets for the "T" and "P" signals have a retentive memory component. Both the "T signal" and the "P signal" CAN be reset with the signal still present at any time in this transient regardless of Containment pressure.

Answer A is incorrect. Both signals CAN be reset at the current Containment Pressure.

Answer C is incorrect. Both signals CAN be reset at the current Containment Pressure.

Answer D is incorrect. Both signals CAN be reset at the current Containment Pressure.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse functional diagram 1-NHY-509048. Lesson SBK LOP L8057I, Integrated Safeguards.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8057I 08RO (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ______

Comments:

Question 56

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	001K5.65	
	Importance Rating	3.2	3.6

K/A 001 = Control Rod Drive System
K5 Knowledge of the following operational implications as they apply to the CRDS:
K5.65; CRDS circuitry, including effects of primary/secondary power mismatch on rod motion.

Proposed Question:

The plant was operating at 100% power when Power Range channel N43 rapidly failed HIGH.

Which of the following describes the expected response of the rod control system?

A. Rods will move OUT due to the power mismatch, then STOP when the power mismatch decays away.

B. Rods will move IN due to the power mismatch, then STOP when the power mismatch decays away.

C. Rods will move IN due to the power mismatch, then move back OUT as the temperature error develops and power mismatch decays away.

D. Rods will move OUT due to the power mismatch, then move back IN as the temperature error develops and power mismatch decays away.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Rods will move IN due to the power mismatch, then STOP when the power mismatch decays away. Rods will not step back out to correct the temperature mismatch because the failed channel now indicates power greater than 103% and will instate the C-2 auto rod withdrawal block.

Answer A is incorrect. The power mismatch circuit will direct inward rod motion.

Answer C is incorrect. Rods will not step back out to correct the temperature mismatch because the failed channel now indicates power greater than 103% and will instate the C-2 auto rod withdrawal block.

Answer D is incorrect. The power mismatch circuit will direct inward rod motion.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse functional drawing 1-NHY-509049.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1406I 03 RO (As available)

Question Source: Bank # 28076
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments:

Question 57

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	002K6.12	
	Importance Rating	3.0	3.5

K/A 002 = Reactor Coolant system
K6 Knowledge of the effect or a loss or malfunction on the following RCS components:
K6.12 Code Safety valves

Proposed Question

The following plant conditions exist:

- The plant is at 100% power
- A notification from the vender has been received identifying that all three Pressurizer Code Safety Valves are INOPERABLE because they may lift at a pressure above their design setpoint.

What is the potential impact of this condition on plant operation?

A. No effect on plant protection for the maximum surge that results from a complete loss of load provided that both PORVs are OPERABLE in AUTOMATIC and Steam Dumps are available.

B. The plant may pressurize above it's Safety Limit without at least a single safety. Each Safety is adequate to relieve the overpressure condition which could occur from a design basis load rejection.

C. No effect on plant protection for the overpressure condition which could occur from a design basis load rejection provided that both PORVs are OPERABLE in AUTOMATIC and rod control is maintained in AUTOMATIC.

D. The RCS may pressurize above it's Safety Limit for the maximum surge that results from a complete loss of load, assuming there is no reactor trip until the first reactor trip system setpoint is reached (no credit taken for direct reactor trip on turbine trip).

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The bases for TS 3.4.4.2 states that all three Pressurizer Code Safety Valves provide protection from a complete loss of load assuming no reactor trip until the first reactor trip system setpoint is reached.

Answer A is incorrect. The bases for TS 3.4.4.4 for PORVs describes their function as related to ability to handle step load decreases in conjunction with the Condenser Steam Dumps, but the PORVs are not a substitute for the Code Safeties.

Answer B is in correct. The bases for TS 3.4.4.2 states that a single Code safety can prevent any RCS overpressure condition which could occur during SHUTDOWN. All three safety valves are required for protection during power operation.

Answer C is incorrect. The bases for the PORVs describe their function as minimizing the undesirable opening of the Code Safeties, but they are not credited as substituting for the Code Safeties.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Technical Specification bases for 3.4.4.2, Safety Valves

Proposed references to be provided to applicants during examination: None

Learning Objective: L8022I 15RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 58

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	014A1.02	
	Importance Rating	3.2	3.6

K/A 014 = Rod Position Indication System
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RPIS controls, including: A1.02 Control rod position indication on control room panels

Proposed Question:

During a reactor startup with no failures, the following alarms are initially present:

- D7730, "One Rod On Bottom"
- D7749, "Two Or More Rods On Bottom"

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

A. When all shutdown rods are fully withdrawn and the Bank Overlap Unit is manually reset.

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

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

D. When all control bank 'D' rods are above the rod insertion limit and the Bank Overlap Unit is manually reset.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. D7730 actuates when any rod in a withdrawn bank is on the bottom. DRPI "defines" a bank withdrawn as greater than 12 steps. D7749 actuates when at least 2 rods are less than 12 steps with the rod bottom bypass de-energized. This will reset when control Bank A reaches the 12 step point.

Answer A is incorrect. The Bank Overlap Unit is reset prior to starting the reactor startup, but these alarms will only reset on actual rod heights greater than setpoint. Additionally, the alarms are based on DRPI indications, not Group Step counter demands.

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

Answer D is incorrect. Resetting the Bank Overlap Unit prior to starting the reactor start up is required to correctly determine rod heights using GRPI, but the alarms are driven from DRPI indication. Additionally, the Shutdown rods also would have to be withdrawn in order to reset these alarms.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8032I, Rod Position Indication.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8032I 08 RO (As available)

Question Source: Bank # 16012
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5
55.43 ______

Comments:

Question 59

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	015K1.04	
	Importance Rating	3.5	3.5

K/A 015 = Nuclear Instrumentation System
K1 Knowledge of the physical connections and/or cause effect relationships between the NIS and the following systems: K1.04 ESF

Proposed Question:

The following plant conditions exist:

- Power Range channel N44 was removed from service and its associated bistables tripped by I&C
- The "D" RCP has developed a seal problem.
- The crew has just performed a downpower to 47% to remove the "D" RCP from service.
- During the downpower Power Range channel N41 began to drift and only decreased to 52%.

What is the impact to Reactor trip protection of this failure?

A. Reactor will trip when the "D" RCP is removed from service because the P-7, At Power permissive, DID NOT reset.

B. The Reactor will trip when the "D" RCP is removed from service because the P-8, Loss of Flow permissive, DID NOT reset.

C. The Reactor will NOT trip when the "D" RCP is removed from service because the P-7, At Power permissive, DID reset

D. The Reactor will NOT trip when the "D" RCP is removed from service because the P-8, Loss of Flow permissive, DID reset.

Answer: B

Explanation (Optional):

Answer B is correct. Reactor will trip when one RCP is removed from service because when N44 is placed in TRIP and N41 stays above 52% the plant will not get 3 of 4 channels less than the 50% P-8, loss of flow for a single loop, setpoint.

Answer A is incorrect. The Reactor will trip, but not because P-7 changed states. The trip condition that would apply in that case would be loss of flow in TWO Reactor Coolant loops.

Answer C is incorrect. If P-7 did change states then the RCS low flow reactor trips would be no longer be enabled.

Answer D is incorrect. If P-8 did change reset then it would require 2 of 4 RCS loops to loose flow in order to trip.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
NI Permissive and Blocks Westinghouse functional drawing 1-NHY-509044. Lesson SBK LOP L8056I, Reactor Protection System.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8056I 19RO and 22RO (As available)

Question Source: Bank # ____
Modified Bank # 22037 (Note changes or attach parent)
New ____

TEB 22037

The following plant conditions exist:

- Power Range channel N44 was removed from service and its associated bistables tripped by I&C, in accordance with OS1211.04, "Power Range NI Instrument Failure".
- Due to a seal problem, the crew has just performed a downpower to 47% and removed the "D" RCP from service.
- Power Range channel N41 begins to drift and is currently reading 52%.

Which of the following should be subsequently performed by the crew as directed by the Unit Supervisor?

A. Verify reactor trip and enter E-0, "Reactor Trip of Safety Injection".

B. Continue with the power reduction and notify I&C to check power range channel N41.

C. Perform calorimetric calculations to correct N41 within 2 hours or be in HOT STANDBY within 6 hours.

D. Declare power range channel N41 inoperable and within one hour make preparations to be in MODE 3 within the next 6 hours.

Answer: A

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Modified from bank

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

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

10 CFR Part 55 Content: 55.41 41.2 to 41.9
55.43 ____

Comments:

During a preliminary review of this question by a NRC examiner it was identified that the K/A match may not be clear. The question appeared to make clearer ties to the K/A linking Nuclear Instrumentation to the Reactor Protection System. The Training material at Seabrook presents the SSPS system, including interactions between RPS and ESFAS, in a single lesson; SBL LOP L8056I, Reactor Protection System.

Specific learning objectives in this lesson include;

- L8056I 12RO; State the ESF signals (bistable inputs to RPS) that are "energize to actuate".
- L8056I 13RO; State the RPS components monitored to provide reactor trip and safeguards actuation bistable status.
- L8056I 18RO; Given any reactor trip of safeguards actuation signal state an accident or unsafe condition the trip is designed to provide protection against.
- L8056I 19RO; Describe eight ESF actuation signals including setpoint and coincidence.
- L8056I 21RO; Describe ten permissives, including setpoint and coincidence.
- L8056I 22RO; Describe eight control permissives, including setpoint and coincidence.

For Seabrook no other link between the Nuclear Instrumentation System and the ESFAS system exists.

Question 60

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	017K4.01	
	Importance Rating	3.4	3.7

K/A 017 = In-core temperature monitoring System
K 4 Knowledge of the ITM system design feature(s) and/or interlock(s) which provide for the following: K4.01 = Input to Subcooling Monitors

Proposed Question:

Which of the following instruments provide input to Train "A" of the Subcooling Monitor?

A. RCS Wide Range Pressure Instrument PT-403 and the average of all Core Exit Thermocouples.

B. RCS Wide Range Pressure Instrument PT-405 and the average of all Core Exit Thermocouples.

C. RCS Wide Range Pressure Instrument PT-405 and the auctioneered highest quadrant average Core Exit temperature.

D. RCS Wide Range Pressure Instrument PT-403 and the auctioneered highest quadrant average Core Exit temperature.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. RCS Wide Range Pressure Instrument PT-405 provides the pressure input to the "A" train instrumentation. The temperature input uses the auctioneered highest quadrant average Core Exit temperature.

Answer A is incorrect. PT-403 is the "B" train pressure instrument. The temperature input uses the auctioneered highest quadrant average temperature not the average of all Core Exit Thermocouples.

Answer B is incorrect. The temperature input uses the auctioneered highest quadrant average Core Exit temperature.

Answer D is incorrect. PT-403 is the "B" train pressure instrument.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8058I, Accident Monitoring.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8058I 13RO (As available)

Question Source: Bank # 23190
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.2
55.43 ______

Comments:

Question 61

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	033A2.03	
	Importance Rating	3.1	3.5

K/A 033 = Spent Fuel Pool Cooling
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the Spent Fuel Pool Cooling System; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations:
A2.03 Abnormal spent fuel pool water level or loss of water level

Proposed Question:

The following plant conditions exist:

- The plant is in Mode 6.
- The Cask Handling Area of the Spent Fuel Pool has been dewatered for Upender repairs
- A failure in the spent fuel pool gate has occurred.
- The Crew has entered OS1215.07, "Loss of Spent Fuel Pool Cooling or Level".

Which of the following Spent Fuel Pool emergency makeup sources has FIRST priority in providing makeup flow to the Spent Fuel Pool if it is rapidly losing inventory?

A. Fire Protection.
B. Condensate Storage Tank.
C. Spent Fuel Pool Purification.
D. Refueling Water Storage Tank.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The Refueling Water Storage Tank is the preferred source of makeup water.

Answer A is incorrect. The Fire Protection system is the third choice.

Answer B is incorrect. The Condensate Storage Tank is the second choice.

Answer C is incorrect. Normal makeup from the CVCS system through the Spent Fuel Pool Purification is not an EMERGENCY makeup source, it is the normal makeup source.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1215.07, Loss of Spent Fuel Pool Cooling or Level.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1192I 07RO and L1192I 08RO (As available)

Question Source: Bank # 16373
Modified Bank # _______ (Note changes or attach parent)
New _______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 62

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	035A1.01	
	Importance Rating	3.6	3.8

K/A 035 = Steam generator system
A1 Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the S/GS controls including: A1.01 S/G wide and narrow range level during startup, shutdown, and normal operation

Proposed Question:

The following plant conditions exist:

- Reactor power is 12%.
- A feed station watch has been established to control SG levels.
- The feed station watch is shifting from the Feed Regulating Valve Bypass Valves to the Main Feed Regulating Valves.
- Steam generator pressures are 1100 psig.
- Main Feed header pressure is 1180 psig.
- Narrow range level in the "B" Steam Generator is 55% and begins to increase.
- Narrow range level in the other 3 Steam Generators is 51% and stable.

Which of the following actions should be taken by the operator to control Steam Generator Levels in accordance with OS1035.12, "Miscellaneous Feedwater Operations"?

A. Lower feed water flow to the "B" steam generator until wide range indicated level begins to turn.

B. Lower feed water flow to the "B" steam generators until narrow range indicated level begins to turn.

C. Decrease Main Feed pump speed to return FW pump DP to program lowering level in the "B" steam generator.

D. Indicated feed flow should be reduced to less than indicated steam flow in the "B" steam generator to stabilize level.

Proposed Answer: A

Which of the following actions should be first taken by the operator to control feedwater?

A. Decrease Main Feed pump speed .

B. Lower feedwater flow to the "A" steam generator until narrow range begins to turn.

C. Lower feedwater flow to the "A" steam generator until wide range begins to turn.

D. Reset Feedwater Isolation and slowly restore feedwater flow to all steam generators.

Answer _C_

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.5
	55.43	____

Comments:

Question 63

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	045K1.20	
	Importance Rating	3.4	3.6

045 = Main Turbine Generator system
Comments: K1 Knowledge of the physical connections and/or cause effect relationships between the MT/G system and the following systems: K1.20 Protection system

Proposed Question:

Which of the following statements correctly describes the Main Turbine Trip System?

A. The 24VDC bus and the 125VDC bus both de-energize to cause a turbine trip. BOTH busses are required to trip in order to cause a turbine trip. This prevents spurious turbine trips in the event one bus fails.

B. The 24VDC bus and the 125VDC bus both de-energize to cause a turbine trip. EITHER bus tripping can cause a trip. Two different power sources provide diverse trip protection.

C. The 24VDC bus will de-energize to cause a turbine trip. The 125VDC bus will energize to cause a turbine trip. EITHER bus tripping can cause a trip. This provides diverse trip protection.

D. The 24VDC bus and the 125VDC bus both energize to cause a turbine trip. BOTH busses tripping are required to cause a trip. A trip on either bus sends a trip to the opposite bus. This feature prevents failures in one trip bus from preventing a turbine trip.

Proposed Answer: C

Explanation (Optional):

Answer C is incorrect. The 24VDC bus de-energizes to trip, and the 125VDC bus energizes to trip. EITHER bus tripping can cause a trip.

Answer A is incorrect. The 125VDC bus energizes to trip. Either bus can cause a trip.

Answer B is incorrect. The 125VDC bus energizes to trip.

Answer D is incorrect. The 24VDC bus de-energizes to trip. Either bus can cause a trip. A trip on either bus does send a trip to the opposite bus and this design was chosen to prevent failures in one trip bus from preventing a turbine trip.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L8049I, Electro Hydraulic Control System.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8049I 19RO (As available)

Question Source:
Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question

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

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

10 CFR Part 55 Content:
55.41 41.2 to 41.9
55.43 ______

Comments:

Question 64

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	056 G 2.1.7	
	Importance Rating	4.4	4.7

K/A 056 = Condensate System
2.1 Conduct of Operations
2.1.7 Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question:

The following plant conditions exist:

- A power increase is in progress.
- The reactor is currently at 75% power.
- CO-P30A and CO-P-30B are in service, with CO-P-30C in standby.
- Main Control Board indications are as follows:
  - Condensate Discharge Header pressure indicates 450 psig.
  - Steam Packing Exhauster flow indicates 21,000 GPM.
  - CO-P-30A current = 500 amps
  - CO-P-30B current = 400 amps
  - VAS alarms for "CONDENSER HOTWELL LEVEL LOW" have annunciated for all hotwells
  - Hotwell level is -9 inches for all hotwells
- Condensate pump "A" trips and condensate pump "C" starts automatically.

Which one of the following describes what happened and the appropriate operator response?

A. Condensate pump "A" tripped on overcurrent, condensate pump "C" started on "A" pump trip, enter OS1231.03, "Turbine Runback/Setback".

B. Condensate pump "A" tripped on low suction pressure, condensate pump "C" started on "A" pump trip, enter OS1231.03, "Turbine Runback/Setback".

C. Condensate pump "A" tripped on overcurrent, condensate pump "C" started on "A" pump trip, enter OS1290.02, "Response to Secondary System Transient ".

D. Condensate pump "A" tripped on low suction pressure, condensate pump "C" started on "A" pump trip, enter OS1290.02, "Response to Secondary System Transient ".

Proposed Answer: C

Explanation (Optional):

Answer C is correct. A standby condensate pump only automatically starts on an electrical trip of a running pump. A setback would also not initiate provided the standby pump started. OS1290.02, "Response to Secondary System Transient " provides the necessary direction to ensure the secondary plant is returned to stable conditions.

Answer A is incorrect. A trip of any condensate pump would not cause a setback provided that the standby pump started.

Answer B is incorrect. There is no direct condensate pump trip on low suction pressure, and given hotwell levels are not low enough to cause a trip.

Answer D is incorrect. There is no direct condensate pump trip on low suction pressure and, there would not be any setback provided the standby condensate pump tripped.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Lesson SBK LOP L1191I, Condensate System. OS1290.02, "Response to Secondary System Transient"

Proposed references to be provided to applicants during examination: None

Learning Objective: L1191I 08RO (As available)

Question Source: Bank # 26720
Modified Bank # (Note changes or attach parent)
New

Question History:
Last NRC Exam Never used on NRC Exam

Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 65

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	072K4.01	
	Importance Rating	3.3	3.6

K/A 072 = Area Radiation Monitoring System
K4 Knowledge of ARM system design feature(s) and/or interlock(s) which provide for the following: K4.01 Containment ventilation isolation

Proposed Question:

The following plant conditions exist:

- Containment on-line purge system is in operation.
- Containment On-Line Purge Radiation Monitors RM-6527A channels 1 & 2 and RM-6527B channels 1 & 2 go into high alarm.

Which of the following describes the automatic response of the system to this condition?

A. All four containment purge supply and exhaust valves close. COP-FN-73, "Containment On Line Purge Supply Fan", continues to run.

B. All four containment purge supply and exhaust valves close and COP-FN-73, "Containment On Line Purge Supply Fan" Trips.

C. One supply and one exhaust valve close. COP-FN-73, "Containment On Line Purge Supply Fan" continues to run.

D. One supply and one exhaust valve close and COP-FN-73, "Containment On Line Purge Supply Fan" Trips.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. RM-6527A 1/2 generates an "A" train CVI which closes COP-V-1 and V-3 and trips the COP fan. RM-6527B 1/2 generates a "B" train CVI which closes COP-V-2 and V-4.

Answer A is incorrect. The train "A" CVI trips COP-FN-73, "Containment On Line Purge Supply Fan".

Answer C is incorrect. Both trains of CVI are actuated so all supply and all exhaust valves close. The train "A" CVI trips COP-FN-73, "Containment On Line Purge Supply Fan".

Answer D is incorrect. Both trains of CVI are actuated so all supply and all exhaust valves close.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Electrical Schematic 1-NHY-310920.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8038I 25RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 ____

Comments:

Question 66

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.4	
	Importance Rating	3.3	3.8

2.1.4 Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, "no-solo" operation, maintenance of active license status, 10CFR55, etc.

Proposed Question:

If a Licensed Control Room Operator has the letters NS (No Solo Operations) printed next to his/her name on the shift schedule this means:

A. He/she cannot be allowed to perform any complex evolutions by himself/herself.

B. He/she must have a SRO licensed supervisor present during all operations that he/she is involved in.

C. He/she must have some other licensed individual in line of sight while performing component manipulation.

D. He/she cannot be allowed to manipulate any controls affecting reactivity or plant power level without a SRO licensed individual present.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The OPMM policy for no solo operations states that he/she must have some other licensed individual in line of sight while performing component manipulation.

Answer A is incorrect. The OPMM policy for no solo operations states that he/she must have some other licensed individual in line of sight while performing ANY component manipulation.

Answer B is incorrect. The OPMM policy for no solo operations states that he/she must have some OTHER LICENSED INDIVIDUAL in line of sight while performing component manipulation. Observation by an SRO is not required.

Answer D is incorrect. The OPMM policy is not restricted to controls involving reactivity or plant power level, it is for ANY component manipulation.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OPMM chapter 1.5, section 5.0 Policies, item 5.3 No Solo Operation, page 1-5.1.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1505I 06RO (As available)

Question Source: Bank # 6633
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam Never used on NRC Exam
Exam Bank History Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.2

Comments:

Question 67

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.23	
	Importance Rating	4.3	4.4

2.1.23 Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question:

The following conditions exist:

- Plant was tripped from 85% power when a rapid power reduction failed to stop a loss of condenser vacuum due to a loss of Circulating Water cooling.
- The Shift Manager wants the crew to shut the MSIVs to protect the main condenser.

What must the Unit Supervisor do before shutting the MSIVs?

A. Get agreement from the crew and invoke 10CFR 50.54(x).

B. Get agreement from the crew and declare a procedure deviation in accordance with OPMM chapter 3 section 1.2.1, "Response to Abnormal or Emergency Conditions."

C. The overriding safety concern is sufficient justification for the Shift Manager or Unit Supervisor to invoke 10CFR50.54(x); crew agreement is not necessary.

D. Since abnormal procedures can be used concurrently with EOPs, no procedure deviation is needed; the crew only needs to say they are using the procedures concurrently.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. Since abnormal procedures can be used concurrently with EOPs, no procedure deviation is needed; the crew only needs to say they are using the procedures concurrently.

Answer A is incorrect. Use of 10CFR 50.54(x) is not required to use AOPs and EOPs in parallel. Further, use of 10CFR50.54(x) can be invoked by the Shift Manager or Unit Supervisor alone and does not require crew concurrence.

Answer B is incorrect. Direction to perform the isolation of the MSIVs is given in the Loss of Condenser Vacuum AOP. A procedure deviation is not required for using AOPs and EOPs in parallel.

Answer C is incorrect. Although 10CFR50.54(x) can be invoked by the Shift Manager or Unit Supervisor alone, direction to perform the isolation of the MSIVs is given in the Loss of Condenser Vacuum AOP so a procedure deviation is not required to use AOPs and EOPs in parallel.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OPMM Chapter 9.2, section 4.7.1, Parallel use of Other Procedures during EOP implementation, and section 4.7.3, procedure compliance policy.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1505I 06 RO and L1513I 07 RO (As available)

Question Source: Bank # 22663
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 68

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.15	
	Importance Rating	2.7	3.4

2.1.15 Knowledge of administrative requirements for temporary management directives, such as standing operating orders, night orders, Operations Memos, etc.

Proposed Question:

Which of the following problems could be addressed as a Standing Operating order?

A. A step in the Turbine Shell and Chest warming procedure does not work as written, so a different action must be substituted.

B. The Turbine Power setpoints of the C-20 AMSAC permissive must be temporarily raised to allow Digital EHC panel work.

C. Direct use of alternate indication to verify containment isolation valve position instead of the Critical Safety Function Status Tree.

D. A procedure change is required during a snowstorm to safely operate a piece of equipment which requires a 10CFR50.59 evaluation within 24 hrs.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. This direction does not violate OPMM chapter 6 for what a SOO can or cannot be used for.

Answer A is incorrect. Standing Operating Orders cannot be used in lieu of changing a SORC approved procedure.

Answer B is incorrect. Standing Operating Orders cannot be used to change a SORC approved setpoint.

Answer D is incorrect. Standing Operating Orders cannot be used to bypass a SORC approved procedure. Each Standing Operating Order also MUST be evaluated for 10CFR50.59 applicability.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OPMM, Chapter 6, section 1.2.1, page 12.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1505I 06RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 ____

Comments:

Question 69

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.1	
	Importance Rating	4.5	4.4

2.2.1 Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity

Proposed Question:

The following plant conditions exist:

- The Plant is completing a Refueling Outage
- A precaution in OS1000.02, "Plant Startup From Hot Standby To Minimum Load", states that the differential boron concentration between the Pressurizer and the Reactor Coolant Loops shall be maintained less than 50 ppm.

Which of the following statements describes the procedural method used to accomplish this?

A. Initial criticality following a Refueling outage is accomplished by continuous dilution to criticality. This provides more even mixing of RCS boron concentration.

B. At least one group of Pressurizer Backup heaters are energized with the Master Pressure Controller in Auto. This causes the Spray valves to partially open.

C. The "C" Reactor Coolant pump is always maintained in service during plant startup activities. The "C" RCP provides the best Pressurizer spray flow.

D. Both trains of RHR are placed in service during plant heatup. This ensures even mixing of the RCS.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Boron concentration is maintained equal by forcing pressurizer sprays. This is the method utilized by OS1000.02.

Answer A is incorrect. Initial criticalities are often done using continuous dilution but this does not enhance or hinder more even mixing of RCS boron concentration.

Answer C is incorrect. The "C" RCP does provide the best Pressurizer spray flow but all Reactor Coolant pump must be running for entry into Mode 2 during plant startup activities in accordance with TS 3.4.1.1.

Answer D is incorrect. In mode 3 RHR may be run on recirculation to equalize boron concentration IN THE RHR loops for the NEXT shutdown, but this does not affect RCS boron concentration.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
MPE OS1000.02, " Plant Startup From Hot Standby To Minimum Load".

Proposed references to be provided to applicants during examination: None

Learning Objective: L1168I 02RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5, 41.10
55.43 43.5, 43.6

Comments:

Question 70

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.20	
	Importance Rating	2.6	3.8

Comments: 2.2.20 Knowledge of the process for managing troubleshooting activities.

Proposed Question:

Which of the following types of Maintenance activities is considered Non-Intrusive troubleshooting?

A. Voltmeter Readings taken from slide links.

B. Clamp on ammeter readings in a MCC bucket.

C. Installation of a Temporary Pressure gauge to observe a pump's discharge pressure.

D. Connection of a MT& E Chart Recorder to allow monitoring a parameter for multiple shifts.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The SSMA defines use of a clamp on ammeter as an example of non-intrusive testing.

Answer A is incorrect. This requires physical connection to an electrical circuit and the SSMA identifies this as intrusive testing.

Answer C is incorrect. The use of temporary pressure gauges for troubleshooting purposes provides contact to process fluids and is identified as intrusive.

Answer D is incorrect. The use of a chart recorder requires physical connection to an electrical circuit and the SSMA identifies this as intrusive testing.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Seabrook Station Maintenance manual, SSMA, MA 4.14, Section 4.2 Definitions.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1229I 02RO (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 71

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.40	
	Importance Rating	3.4	4.7

2.2.40 Ability to apply Technical Specifications for a system.

Proposed Question:

The following plant conditions exist:

- The plant is at 75% power and increasing to 100% power. .
- The Unit Supervisor identifies that the plant cannot meet the requirements of a Limiting Condition for Operation (LCO) AND the LCO's associated Action Statement(s).

Which of the following describes the course of action required to be taken by the crew?

A. Operation may continue provided the Surveillance Requirements are satisfied within 25% percent of the specified surveillance interval from the point of discovery.

B. Immediately stop the plant power ascension until the component can be restored to OPERABLE status, not to exceed 24 hours or the surveillance interval - whichever is greater.

C. Immediately initiate actions to place the unit in a MODE in which specification does not apply by placing it in at least HOT STANDBY within the next 6 hours, at least HOT SHUTDOWN within the following 6 hours, and at least COLD SHUTDOWN within the subsequent 24 hours.

D. Within 1 hour action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it in at least HOT STANDBY within the next 6 hours, at least HOT SHUTDOWN within the following 6 hours, and at least COLD SHUTDOWN within the subsequent 24 hours.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The situation described would put the unit in TS 3.0.3. This answer describes TS 3.0.3.

Answer A is incorrect. Technical Specification 4.0.2. provides a 25% maximum allowable extension but, the plant would have to be in a condition described within the applicable action statements in order to apply TS 4.0.2.

Answer B is incorrect. The LCO and action statements apply for all of Mode 1 conditions, regardless of plant power ascension or stable conditions. Stopping the power ascension would not change any conditions.

Answer C is incorrect. TS 3.0.3 allows 1 hour of preparation time before beginning the plant shutdown.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Technical Specification section 3/4.0 Applicability.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8010I 05 (As available)

Question Source: Bank # 25831
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.2, 43.5

Comments:

Question 72

Examination Outline Cross-Reference:

	RO	SRO
Level	RO	SRO
Tier #	3	
Group #	3	
K/A #	2.3.4	
Importance Rating	3.2	3.7

2.3.4 Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question:

The following plant conditions exist:

- A Site Area Emergency has been declared based on EAL HS3, Other conditions existing which in the judgment of the STED/SED warrant declaration of Site Area Emergency.
- The plant is at 100% power
- The Cask Handling Crane failed while lifting a fully loaded Dry cask from the cask handling platform.
- The Cask was loaded with fuel but was not sealed yet.
- The Cask fell back onto the platform and partially tipped against the Spent Fuel Pool.
- Airborne radiation readings are normal.
- The TSC had determined that the fuel is not in danger of overheating.
- A team of riggers are being sent into the Spent Fuel Building to right the cask.

The OSC Team leader's current TEDE dose is as follows:

- Exposure for the year: 350 mrem
- Current total lifetime exposure: 1100 mrem
- Current age: 37 yrs

Based on these conditions, what is the maximum emergency exposure this Team Leader may receive while performing this action?

A. 3900 mrem TEDE

B. 10000 mrem TEDE

C. 11100 mrem TEDE

D. 25000 mrem TEDE

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The emergency exposure limits for protecting valuable equipment is 10 Rem.

Answer A is incorrect. This distractor is based on the workers maximum federal dose limit (5 Rem) minus his current lifetime exposure of 1100 mrem.

Answer C is incorrect. This distractor is based on the emergency exposure limits for protecting valuable equipment of 10 Rem plus his current lifetime exposure of 1100 mrem.

Answer D is incorrect. This distractor is based on the emergency dose limit for protection of valuable property.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ER 4.3, pg 5-7, figure 2 and 3

Proposed references to be provided to applicants during examination: None

Learning Objective: L1525I13RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.12/13
55.43 43.4

Comments:

Question 73

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.13	
	Importance Rating	3.4	3.8

2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:

The following plant conditions exist:

- The plant is in Mode 6
- The PSO is briefing an NSO to re-align a Refueling Cavity Purification filter to provide additional Cavity cleanup.
- All components to be re-aligned are located inside the Bioshield under the "D" RCP.
- It is conservatively expected to take 30 minutes to perform the evolution.
- The General Area dose Rate at the filter is 112 mrem/hr
- There is a dose reading of 750 mrem/hr at 30 cm on the underside of the filter.
- The General Area dose rates inside the Bioshield range from 5 mrem/hr to 500 mrem/hr
- There is a Dose rate reading of 1120 mrem/hr at 30 cm under the "A" RCP.

What is the expected radiation posting and entry requirements for this evolution?

A. This is a Radiation Area. The NSO must be signed on to a RWP, have knowledge of the dose rates, and have a TLD plus Electronic Dosimetry.

B. This is a High Radiation Area. The NSO must be signed on to a RWP, have knowledge of the dose rates, have a TLD plus Electronic Dosimetry.

C. This is a Technical Specification Locked High Radiation Area. The NSO must be signed on to a RWP, have knowledge of the dose rates, have a TLD plus Electronic Dosimetry, and have specific authorization to enter a Locked High Radiation area.

D. This is a Very High Radiation Area. The NSO must be signed on to a RWP, have knowledge of the dose rates, have a TLD plus Electronic Dosimetry, and have specific authorization to enter a Very High Radiation area.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The 1120 mrem/hr Radiation field from the RCP exceeds 1000 mrem/hr which makes this is a Technical Specification Locked High Radiation Area. The entry requirements for a TS Locked High Radiation Area are given.

Answer A is incorrect. The 1120 mrem/hr Radiation field from the RCP exceeds 1000 mrem/hr which makes this is a Technical Specification Locked High Radiation Area.

Answer B is incorrect. The 1120 mrem/hr Radiation field from the RCP exceeds 1000 mrem/hr which makes this is a Technical Specification Locked High Radiation Area.

Answer D is incorrect. Radiation levels does not exceed the 500 Rem/hr at 1 meter which would constitute a Very High Radiation Area.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
RP 9.1 FIGURE 5.2, Requirements for Entry into Radiological Areas and SSRP chapter 1-3, pages 1-3.15, 1-3.16

Proposed references to be provided to applicants during examination: None

Learning Objective: L1229I 03RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.12
55.43 43.4

Comments:

Question 74

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.28	
	Importance Rating	3.2	4.1

2.4.28 Knowledge of procedures relating to a security event (non-safeguards information)

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- The Security Shift Supervisor informs the Control Room that armed intruders and security are exchanging fire in the vicinity of the Emergency Feedwater Pumphouse.
- The control room has been notified that a security code RED event is in progress.

Which of the following is directed by OS1290.03, “Response to a Security Event”?

A. The Shift Manager invokes 10 CFR 50.54(x). Both Emergency Diesel Generators are started and the SUFP is started to provide EFW flow.

B. Trip the Reactor, Go to E-0, “Reactor Trip or Safety Injection” and when Immediate Actions are complete trip B and D Reactor Coolant Pumps.

C. The Shift Manager and Security Supervisor determine if the Threat is Credible. Notify station personnel that a two-person line-of site rule has been established.

D. Close and Dog the Control Room Tornado Door. Sound the Plant Evacuation Alarm and announce that “A Security Event is in Progress. All Personnel should Exit Vital Areas. Evacuate the Protected Area to the Assembly Area”.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. OS1290.03, "Response to a Security Event" requires these actions when a Code Red is declared

Answer A is incorrect. OS1290.03 does not direct starting Emergency Diesel Generators or starting the SUFP. These actions also could be taken without violating TS so invoking 10CFR50.54x would not be required.

Answer C is incorrect. In this case the Shift Manager AND Security Supervisor do not have to determine the Threat is Credible because Security has notified the crew than an event is in progress. Also the two-person line-of site rule is established during a code Yellow response to insider threat, not a Code Red.

Answer D is incorrect. During a code Red the Control Room Tornado Door would be dogged and closed but no movement of plant personnel is allowed.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1290.03, "Response to a Security Event".

Proposed references to be provided to applicants during examination: None

Learning Objective: L1174I 02SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 75

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.41	
	Importance Rating	2.9	4.6

2.4 Emergency Procedures / Plan
2.4.41 Knowledge of the emergency action level thresholds and classifications.

Proposed Question:

What is the time limit to make an emergency plan declaration based on Critical Safety Function Status tree indications?

A. The Classification must be determined and announced to the Crew within 15 minutes of the status tree display being validated.

B. The Classification must be determined and communicated to the NRC within 15 minutes of the status tree display being validated.

C. The Classification must be determined and communicated to the States within 15 minutes of the status tree display being validated.

D. The Classification must be determined within 15 minutes of the status tree display being validated. An additional 15 minutes is allowed to announce the classification to the Crew.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. ER 1.1 states that critical safety functions must be verified as valid before using them to classify an event. The procedure further states that the classification must be made within 15 minutes of clearly receiving indications that an EAL is exceeded and, the classification is considered made when it is announced to the crew as an UPDATE.

Answer B is incorrect. The NRC notification MUST be made within 1 hr of the initial classification (when the announcement is made).

Answer C is incorrect. An additional 15 minutes is allotted from the time of classification until the time the states are notified.

Answer D is incorrect. The event must be classified within 15 minutes of being validated. An additional 15 minutes is not allotted to make the announcement to the crew.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ER 1.1, Section 1.1 Discussion

Proposed references to be provided to applicants during examination: None

Learning Objective: L1509I 23RO (As available)

Question Source:
Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content:
55.41 41.10
55.43 43.5

Comments:

U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information

Name:	
Date:	Facility/Unit: SEABROOK STATION
Region: I ☒ II ☐ III ☐ IV ☐	Reactor Type: W ☒ CE ☐ BW ☐ GE ☐
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

All work done on this examination is my own. I have neither given nor received aid.

____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	75 / 25 / 100	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

Question 76, SRO 1

Examination Outline Cross-Reference:

Level	RO	SRO
Tier #		1
Group #		1
K/A #	022 G 2.4.45	
Importance Rating		4.3

022 Loss of Reactor Coolant makeup
2.4 Emergency Procedures / Plan
2.4.45 Ability to prioritize and interpret the significance of each annunciator or alarm

Proposed Question:

The following plant conditions exist:

- A Tube leak has been identified on the "B" Steam Generator.
- The crew has entered OS1227.02, "Steam Generator Tube Leak".
- The leak size has been estimated as 90 gpm.
- The crew is preparing for the power reduction and plant shutdown.
- The PSO is maintaining VCT level at 40% by performing a blended makeup at 90 gpm through CS-FCV-110B, "Boric Acid Blender to CHG Pumps".
- The Control Board Monitor notes the following alarms:
  - VAS alarm "D4670: BORIC ACID BLEND DEVIATION HI/LO".
  - VAS alarm "D4665: VCT LEVEL LOW (185)".
  - VAS alarm "D4669: VCT LEVEL LOW (112)".

What crew actions are required based on these conditions?

A. Adjust blended makeup flowrate to determine if VCT level and Plant power can be maintained.

B. CS-FCV-110B, "Boric Acid Blender to CHG Pumps", has automatically CLOSED. Placing CS-FCV-111B, "Boric Acid Blender to VCT", in AUTO/OPEN will allow the blended makeup flow to restart.

C. OPEN RWST Suction Valves CS-LCV-112D and CS-LCV-112E, CLOSE CS-LCV-112B and CS-LCV-112C, THEN trip the Reactor and go to E-0, "Reactor Trip or Safety Injection".

D. Trip the Reactor. When Reactor Trip is verified THEN actuate Safety Injection. Go to E-0, "Reactor Trip or Safety Injection".

Proposed Answer: C

Explanation (Optional):

Answer C is correct. VCT level can no longer be maintained. In that event the procedure directs swapping CS pump suction to the RWST, THEN tripping the Reactor and going to E-0, "Reactor Trip or Safety Injection".

Answer A is incorrect. The blended make up terminated when the Boric Acid Flow deviation occurs.

Answer B is incorrect. CS-FCV-111B will also auto close on the flow deviation alarm. Also, 75 gpm is the maximum flow rate allowed to the top of the VCT.

Answer D is incorrect. For a 90 gpm leak the preferred inventory control method would be to start a second charging pump if required, but not initiate a SI unless it was clearly needed.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
VAS procedure VPRO D4670. OS1227.02, step 2 and step 4.

Proposed references to be provided to applicants during examination: None

Learning Objective: ________________ (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5, 43.6

Comments:

Question 77, SRO 2

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	026 G 2.4.4	
	Importance Rating		4.7

026 = Loss of Component Cooling Water
2.4 Emergency Procedures / Plan
2.4.4. Ability to recognize abnormal indications for system operating parameters that are entry level conditions for emergency and abnormal operating procedures.

Proposed Question:

The following conditions exist:

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

  "D4266: RCP A THERM BARRIER COOLING FLOW HI".
- Pressurizer level is slowly decreasing.
- VCT level is slowly decreasing.
- Charging flow is increasing.

Which of the following procedures should be entered by the Unit Supervisor to address this condition?

A. OS1201.02, "RCS Leak" to isolate the RCS leak.

B. OS1201.01, "RCP Malfunction" to evaluate continued RCP operation.

C. OS1202.02, "Charging System Failure", to restore VCT and Pressurizer level.

D. OS1212.01, "PCCW System Malfunction" to isolate Thermal Barrier cooling to the "A" RCP.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The conditions given describe an RCS to Thermal Barrier system leak. OS1212.01, "PCCW System Malfunction" will provide direction to isolate the Thermal Barrier cooling to the heat exchanger.

Answer A is incorrect. The conditions given describe an RCS to Thermal Barrier system leak. OS1201.02, "RCS Leak" does not provide direction to isolate the Thermal Barrier cooling to the heat exchanger to isolate the RCS leak.

Answer B is incorrect. The conditions given describe an RCS to Thermal Barrier system leak. OS1201.01, "RCP Malfunction" does not provide direction to isolate the Thermal Barrier cooling to the heat exchanger to isolate the RCS leak.

Answer C is incorrect. The conditions given describe an RCS to Thermal Barrier system leak. Charging system, Pressurizer Level and VCT level are changing in response to the Thermal Barrier system leak. There is no failure in the charging system.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1212.01, Primary Component Cooling Water System Malfunction.

Proposed references to be provided to applicants during examination: None

Learning Objective: L3157I 03SR (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 78, SRO 3

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	038EA2.12	
	Importance Rating		4.2

038 = Steam Generator Tube Rupture
EA2 Ability to determine or interpret the following as they apply to a SGTR: EA2.12 Status of MSIV activating system

Proposed Question:

The following plant conditions exist:

- The "B" Steam Generator has ruptured.
- An RCS cooldown is in progress using the condenser steam dumps.
- The Unit Supervisor has directed the PSO to perform E-3 Attachment B, "Blocking Low Steam Line Pressure SI".

What is the possible consequence, and procedural transition implications, of the RO failing to block the Low Steamline Pressure SI during the cooldown?

A. A Main Steam Line Isolation will actuate on High Rate of Steam Generator pressure decrease during the plant cooldown. The Post SGTR cooldown can be accomplished using ES-3.2, "Post-SGTR Cooldown using Backfill".

B. A Safety Injection on Low Steamline pressure will actuate as the Steam Generators are depressurized during the plant cooldown. This will require a transition to ECA-3.1, "SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired".

C. A Safety Injection on Pressurizer Low pressure will actuate as RCS pressure decreases from the cooldown. This will require a transition to ECA-3.1, "SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired".

D. A Main Steam Line Isolation (MSI) will actuate as the Steam Generators are depressurized during the plant cooldown. The Post SGTR cooldown can be accomplished using ES-3.2, "Post-SGTR Cooldown using Backfill".

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The Low steam line Main Steam Line Isolation will actuate when steam header pressure decreases below the actuation setpoint slowing down the E-3, "Steam Generator Tube Rupture", cooldown. Use of ES-3.2, "Post-SGTR Cooldown using Backfill" would preclude the need for using the condenser hotwells so this could be used for the post SGTR recovery.

Answer A is incorrect. A Main Steam Line isolation on high rate will not occur until after the P-11 block is accomplished.

Answer B is incorrect. If ECCS re-initiation is required after ECCS pumps have been stopped and placed in standby then a transition to ECA-3.1 would be made after the pumps are manually started. Given this is a tube rupture event the Safety Injection is already reset at this point of the procedure and no further automatic SI will actuate.

Answer C is incorrect. If ECCS re-initiation is required after ECCS pumps have been stopped and placed in standby then a transition to ECA-3.1 would be made after the pumps are manually started. Given this is a tube rupture event the Safety Injection is already reset at this point of the procedure and no further automatic SI will actuate.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Westinghouse Functional 1-NHY-509048.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1205I 03 RO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 ____
55.43 43.5

Comments:

Question 79, SRO 4

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	062 G 2.2.25	
	Importance Rating		4.2

062 = Loss of Nuclear Service Water
2.2 Equipment Control
2.2.25 Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.

Proposed Question:

The following plant conditions exist:

- 1-SW-P-41D is in Pull-To-Lock (PTL) to allow external painting of the pump casing.
- 1-SW-P-41B breaker tripped on Overcurrent.
- The Crew has entered OS1216.01, "Degraded Ultimate Heat Sink" due to a loss of all train "B" Ocean SW pumps.
- The train "B" Tower Actuation (TA) signal did NOT actuate and was manually actuated.
- OS1216.01, "Degraded Ultimate Heat Sink", directs placing both Ocean SW pump control switches in PTL and the affected Cooling Tower Pump control switch in NA-START.

What are the Technical Specification implications of this alignment?

A. Both the Ocean SW loop and Cooling Tower loop are OPERABLE.

B. The Ocean SW loop is INOPERABLE because SW-P-41B failed. The Cooling Tower Loop is OPERABLE.

C. The Ocean SW loop is INOPERABLE because both SW pumps are in PTL. The Cooling TOWER LOOP is OPERABLE.

D. The Ocean SW loop is INOPERABLE because SW-P-41B failed. The Cooling Tower loop is INOPERABLE because an automatic TA should have occurred.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The Ocean SW loop can remain Operable as long as one Ocean Pump is available. The "D" SW pump has remained available because it never a failure. Placing Both Ocean SW pumps in PTL does not affect operability because manual actions are required to shift from the Cooling Tower to the Ocean. The Cooling Tower loop is OPERABLE because there was no demand for a Tower Actuation after both Ocean SW pump breakers were open.

Answer B is incorrect. The Ocean SW loop can remain Operable as long as one Ocean Pump is available.

Answer C is incorrect. Placing Both Ocean SW pumps in PTL does not affect operability because manual actions are required to shift from the Cooling Tower to the Ocean.

Answer D is incorrect. The Ocean SW loop can remain Operable as long as one Ocean Pump is available. Placing Both Ocean SW pumps in PTL does not affect operability because manual actions are required to shift from the Cooling Tower to the Ocean. The Cooling Tower loop is OPERABLE because there was no demand for a Tower Actuation after both Ocean SW pump breakers were open.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Technical Specification Bases for 3 /4. 7. 4, Service Water Ultimate Heat Sink.

Proposed references to be provided to applicants during examination: None

Learning Objective: L80371 06,12,13 (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5, 41.7
55.43 43.2

Comments:

Question 80, SRO 5

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	W/E05EA2.1	
	Importance Rating		4.4

W/E05 = Loss of Secondary Heat Sink
EA2. Ability to determine and interpret the following as they apply to the (Loss of Secondary Heat Sink): EA2.1 Facility conditions and selection of appropriate procedures during abnormal and emergency operations

Proposed Question:

The following plant conditions exist:

- You are directing response to a LOCA in accordance with E-1, "Loss of Reactor or Secondary Coolant."
- A RED path is noted on the Heat Sink Critical Safety Function Status Tree and you have transitioned to FR-H.1, "Response to Loss of Secondary Heat Sink".
- The Motor Driven EFW Pump failed to start.
- The Steam Driven EFW Pump started but total available EFW flow with all EFW throttle valves OPEN is 350 gpm.
- All SG pressures are 950 psig and STABLE.
- All SG levels are 29% Wide Range and DECREASING.
- RCS pressure is 470 psig and DECREASING
- RCS Hot leg temperatures are 535 deg F and SLOWLY DECREASING.
- Containment pressure is 17 psig and INCREASING.

Which of the following actions should be directed next?

A. Transition back to E-1, "Loss of Reactor or Secondary Coolant". RCS Pressure is below intact SG pressure.

B. Transition back to E-1, "Loss of Reactor or Secondary Coolant". Total Feed flow is less than 500 GPM due to Operator actions taken in E-1.

C. Remain in FR-H.1, "Response to Loss of Secondary Heat Sink and establish EFW flow greater than 500 gpm from SUFP to one intact DRY Steam Generator.

D. Remain in FR-H.1, "Response to Loss of Secondary Heat Sink", and Immediately perform Steps 10 – 14, to initiate Feed and Bleed because 3 Steam Generators levels are less than 51% WR.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. A Transition back to E-1, "Loss of Reactor or Secondary Coolant" is directed if RCS Pressure is below intact SG pressure because a secondary heat sink is no longer required.

Answer B is incorrect. Total Feed flow was less than 500 GPM but this was not due to Operator actions.

Answer C is incorrect. A Transition back to E-1, "Loss of Reactor or Secondary Coolant" is directed if RCS Pressure is below intact SG pressure because a secondary heat sink is no longer required.

Answer D is incorrect. A Transition back to E-1, "Loss of Reactor or Secondary Coolant" is directed if RCS Pressure is below intact SG pressure because a secondary heat sink is no longer required.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
FR-H.1, "Response to Loss of Secondary Heat Sink", Step 1, 2, the cautions prior to step 1 and prior to step 3, and the OAS page for dry SG feeding criteria.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1211I 03RO (As available)

Question Source: Bank # 26635
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 ______
55.43 43.5

Comments:

Question 81, SRO 6

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	077AA2.07	
	Importance Rating		4.0

077 = Generator Voltage and Electrical Grid Disturbances
AA2. Ability to determine and interpret the following as they apply to Generator Voltage and Electric Grid Disturbances: AA2.07 Operational status of engineered safety features

Proposed Question:

The following plant conditions exists:

- The Plant is at 100% power.
- The crew has entered OS1246.02, "Degraded Vital AC Power (Plant Operating)" in response to a Degraded Grid condition.
- The BOP operator reports that Bus 5 and Bus 6 voltages are 3950 Volts and SLOWLY DECREASING.

Which of the following describes the actions required by procedure OS1246.02 and the reason for these actions as voltage continues to decrease?

A. IF both busses are less than 3933 volts for greater than 15 mins THEN trip the Reactor and transfer both busses to EDG power after immediate actions. The Emergency bus voltage has reached the LOP 2nd level UV protection setpoint so manual action to protect against further degradation is warranted.

B. IF both busses are less than 3600 Volts THEN trip the Reactor and verify both busses have transferred to EDG power during immediate Actions. The Emergency bus voltage has reached the LOP 2nd level UV protection setpoint and automatic protective actions have failed to protect against further degradation.

C. IF both busses are less than 3933 volts for greater than 15 mins THEN trip the Reactor and transfer both busses to RATs after immediate Actions. The Emergency bus voltage has reached the LOP 1st level UV protection setpoint so manual action to protect against further degradation is warranted.

D. IF both busses are less than 3600 Volts THEN trip the Reactor and verify both busses have transferred to RATs after immediate Actions. The Emergency bus voltage has reached the LOP 1st level UV protection setpoint and automatic protective actions have failed to protect against further degradation.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. 3933 volts is the LOP 2nd level UV protection setpoint. This "line in the sand" has been drawn to take manual action before further degradation. There is not any automatic action that would occur at this level without a coincident SI signal. Step 16 of the procedure directs tripping the Reactor and transferring both busses to EDG power after immediate actions rather than leaving the plant in a degraded condition.

Answer B is incorrect. 3600 Volts is a conservative number chosen in the abnormal to take further actions, not the LOP 2nd level UV protection setpoint. Also, although bus voltage would be below the 2nd level UV protection setpoint, there is not a coincident SI signal so no automatic actions should have occurred. This means the busses would not have initiated an auto transfer to the EDGs.

Answer C is incorrect. 3933 volts is the LOP 2nd level UV protection setpoint not the 1st level UV protection setpoint (~2900 volts). Because the offsite power source is degraded the busses would not be transferred to the RATs after immediate Actions, but to the EDGs.

Answer D is incorrect. 3600 Volts is a conservative number chosen in the abnormal to take further actions, not the LOP 1st level UV protection setpoint. Because the offsite power source is degraded the busses would not be transferred to the RATs after immediate Actions. If the buses had reached the LOP 1st level UV protection setpoint then automatic protective actions should have protected against further degradation.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1246.02, "Degraded Vital AC Power (Plant Operating)", step 16.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8013I 15RO, L1199I 15RO, L3039I 10SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 82, SRO 7

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	003 G 2.2.38	
	Importance Rating		4.5

003 = Dropped Control Rod
Comments: 2.2 Equipment Control
2.2.38 Knowledge of conditions and limitations in the facility license.

Proposed Question:

Why does OS1210.05, "Dropped Rod', direct a manual Reactor Trip if more than one control rod has been dropped?

A. Unanalyzed Rod configurations may invalidate the assumed rod drop time and rod worth used in the safety analyses.

B. All corrected values of predicted Moderator Temperature Coefficient may become more positive than the T.S. limits during multiple rod drop events.

C. Xenon oscillations are directly proportional to the amount of time the rods have been dropped and localized power peaking is inversely proportional to the magnitude of the xenon oscillations.

D. Multiple rod drops or partial rod drops beyond those limited variations that allow continued power operation in Technical Specifications may produce power distributions outside of design limits.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The TS bases for 3.1.3, Movable Control Assemblies, states in the second paragraph that "ACTION statements which permit limited variations from the basis requirements are accompanied by additional restrictions which ensure the original design criteria are met... ..In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation." The station specific AOP for dropped rods has determined that operating outside those limited variations that allow continued power operation in Technical Specifications may produce power distributions outside of design limits. The Dropped Rod procedure conservatively trips the plant when it has been determined that multiple dropped rods has occurred because the required safety analysis may not support continued operation and the evaluation could not reasonably be expected to be performed within the 1 hr TS limit to recover the rods.

Answer A is incorrect. Multiple tripped rods would not invalidate any assumed rod drop times.

Answer B is incorrect. The TS bases 3.1.1.3, Moderator Temperature Coefficient, establishes that Control Rod Withdrawal limits ensures that all corrected values of predicted Moderator Temperature Coefficient do not become more positive than the T.S. limits. This is not related to multiple rod drop events.

Answer C is incorrect. The size of Xenon oscillations and localized power peaking factors are related to amount of time allowed in TS to recover a dropped rod but the 2nd paragraph of TS bases 3.1.3, Movable Control Assemblies, states that compliance with localized power peaking factors can only be assured by actual measurement, not by providing limitations on the number of control rods that can be safely recovered if they drop.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
T.S. Bases for 3.1.1.3 Moderator Temperature Coefficient, 3.1.3, Movable Control Rod assemblies. Also: protected references in OS1210.05, Dropped Rod; SS# 44084, Operator response to more than one Dropped rod and SEN 181, Recurring Event, continued power operations after multiple dropped rods place reactor outside analyzed conditions.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1185I 13SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question

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

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

10 CFR Part 55 Content:	55.41	41.7, 41.10
	55.43	43.7

Comments: 2.2 Equipment Control
2.2.38 Knowledge of conditions and limitations in the facility license.

Question 83, SRO 8

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	024 G 2.4.34	
	Importance Rating		4.1

024 = Emergency Boration
2.4 Emergency Procedures / Plan
2.4.34 Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question:

The following conditions exist:

- The plant was operating in Mode 1 at 100% power.
- A fire in the Seismic Monitoring Cabinet has forced an evacuation of the Control Room.
- The Crew is responding to the Remote Safe Shutdown (RSS) Panels.

Which of the following is the prescribed method of ensuring sufficient RCS boration for Cold shutdown in this condition?

A. At the RSS panels shift CS pump suction to the RWST. Start borating using a Boric Acid Transfer Pump and the Emergency Boration valve. Inject Boric Acid required for Cold Shutdown by calculation or sample.

B. Prior to leaving the Control Room start a boration using a Boric Acid Transfer Pump and the Emergency Boration valve. Monitor WR Excore Neutron Flux less than 1.0 E-3% at RSS panel throughout the cooldown to ensure sufficient boration.

C. Prior to leaving the Control Room start a boration using a Boric Acid Transfer Pump and the Emergency Boration valve. Verify sufficient Boric Acid for cold shutdown injected by sample or calculated volume when the RSS panels are manned.

D. At the RSS panels shift CS pump suction to the RWST. Start borating using a Boric Acid Transfer Pump and the Emergency Boration valve. Monitor WR Excore Neutron Flux remains less than 1.0 E-3% at RSS panel throughout the cooldown to ensure sufficient boration.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The remote safe shutdown procedure directs shifting CS pump suction to the RWST, starting a Boric Acid Transfer Pump and opening the Emergency Boration valve. The required amount of Boric acid can be added until it is verified by sample or, if that is not available, by calculated volume.

Answer B is incorrect. The emergency boration flowpath using a Boric Acid Transfer Pump and the Emergency Boration valve is started from the RSS panel, not the main control room. Neutron flux is monitored using the WR Excore Neutron Flux but this is done as a backup to monitor core reactivity level, not the primary method of verifying sufficient boration for Shutdown Margin.

Answer C is incorrect. The emergency boration flowpath using a Boric Acid Transfer Pump and the Emergency Boration valve is started from the RSS panel, not the main control room although it is true that the correct volume of Boric acid required can be added by calculation.

Answer D is incorrect. The remote safe shutdown procedure does direct shifting CS pump suction to the RWST, starting a Boric Acid Transfer Pump and opening the Emergency Boration valve but monitoring Neutron flux using the WR Excore Neutron Flux is done as a backup to monitor core reactivity level, not the primary method of verifying sufficient boration for Shutdown Margin.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1200.02, " Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities". RE-18, "Shutdown Margin Values".

Proposed references to be provided to applicants during examination: None

Learning Objective: L3063I 08 SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.6

Comments:

Question 84, SRO 9

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	068AA2.05	
	Importance Rating		4.3

068 = Control Room Evacuation
AA2. Ability to determine and interpret the following as they apply to the Control Room Evacuation: AA2.05 Availability of heat sink

Proposed Question:

The following plant conditions exist:

- A Fire has been confirmed in the cable spreading room.
- The Crew is responding using OS1200.00, "Response to Fire or Fire Alarm Actuation".
- The Unit Supervisor has ordered all ASDV mode selector switches to CLOSE.
- Control Room Evacuation will be accomplished using OS1200.02, "Safe Shutdown and Cooldown From the Remote Safe Shutdown Facilities".

How will adequate heat removal be provided in OS1200.02 as the crew mans the Remote Safe Shutdown Panels?

A. The Condenser Steam Dumps will MODULATE OPEN in the Tavg mode.

B. EFW flow will be maintained greater than 500 GPM total to all Steam Generators

C. The Condenser Steam Dumps will MODULATE OPEN in the Steam Pressure Mode.

D. Steam header pressure will be allowed to increase to the Steam Generator Safety Valves setpoints.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. The MSIVs have been closed procedurally. The ASDVs will no longer automatically open after they are placed in CLOSE. The RCS temperature will rise until the Steam header pressure increases to the Steam Generator Safety Valves setpoints.

Answer A is incorrect. The Condenser Steam Dumps are isolated due to closure of the MSIVs.

Answer B is incorrect. 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.

Answer C is incorrect. The Condenser Steam Dumps are isolated due to closure of the MSIVs.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1200.00, "Response to Fire or Fire Alarm Actuation", Step 5. OS1200.02, "Safe Shutdown and Cooldown from the Remote Safe Shutdown Facilities", Step 1.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8210I 02RO, L3063I 07SR (As available)

Question Source: Bank # 22643
Modified Bank # ______ (Note changes or attach parent)
New ______

Question History:
Last NRC Exam: Never used on NRC Exam
Exam Bank History: Directly from bank

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

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

10 CFR Part 55 Content: 55.41 ______
55.43 43.1

Comments:

Question 85, SRO 10

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	W/E09 &E10 G2.1.25	
	Importance Rating		4.2

W/E09 = Natural Circulation Operations
W/E10 = Natural Circulation with Steam Void in Vessel with/without RVLIS
2.1 Conduct of Operations
2.1.25 Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question:

The following plant conditions exist:

- A Hurricane is approaching Seabrook Station
- The plant tripped from 100% power due to a Total Loss of Offsite Power.
- The Operators successfully restored power to both Emergency Busses.
- A RCS cooldown of 90 degrees per hour is in progress.
- RCS Temperature is 450 degs F.
- RCS Wide Range pressure is 1000 psig.
- PZR Level is 50% and SLOWLY DECREASING.
- RVLIS full range level is 80% and SLOWLY INCREASING.
- RCP seal injection is isolated and Normal charging and letdown are in service.
- The TSC has recommended that the Cooldown should continue.
- The Crew has transitioned to ES-0.3, "Natural Circulation Cooldown with Steam Void in the Vessel (with RVLIS)" from ES-0.2, "Natural Circulation Cooldown".

Determine what actions should be taken to control RCS pressure given the attached "Figure ES-0.3-1, Pressure Temperature Limits".

A. Subcooling is greater than required. Continue RCS cooldown and depressurization using Auxiliary Spray to maintain RCS subcooling GREATER THAN 60 degs F.

B. Subcooling is greater than required. Continue RCS cooldown and depressurization using one PZR PORV to maintain RCS subcooling GREATER THAN 60 degs F.

C. Subcooling is less than required. Stop RCS cooldown and depressurization. Re-pressurize the RCS to increase RVLIS full range level to GREATER THAN 90%.

D. Subcooling is less than required. Control Charging and Letdown as necessary to INCREASE subcooling and restore PZR level to GREATER THAN 68%.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. The actual RCS subcooling is 96 degrees, which is greater than 60 degrees Subcooling is greater than required. Use of Auxiliary Spray to maintain RCS subcooling GREATER THAN 60 degs F is preferred while continuing RCS cooldown and depressurization.

Answer B is incorrect. If letdown is available auxiliary spray is preferred over use of one PZR PORV to maintain RCS subcooling GREATER THAN 60 degs F.

Answer C is incorrect. Actual subcooling is greater than required. During the cooldown RVLIS full range level is maintained greater than 68% and pressurizer level is cycled between 30% and 90%.

Answer D is incorrect. Actual subcooling is greater than required. During the cooldown RVLIS full range level is maintained greater than 68% and pressurizer level is cycled between 30% and 90%.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ES-0.2, Natural Circulation Cooldown, Step 16. ES-0.3, "Natural Circulation Cooldown with Steam Void in Vessel (With RVLIS), steps 3 thru 5. and Figure ES-0.3-1, Pressure Temperature Limits

Proposed references to be provided to applicants during examination: Figure ES-0.3-1, Pressure Temperature Limits

Learning Objective: L1213I 11RO, L3051I 10SR (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

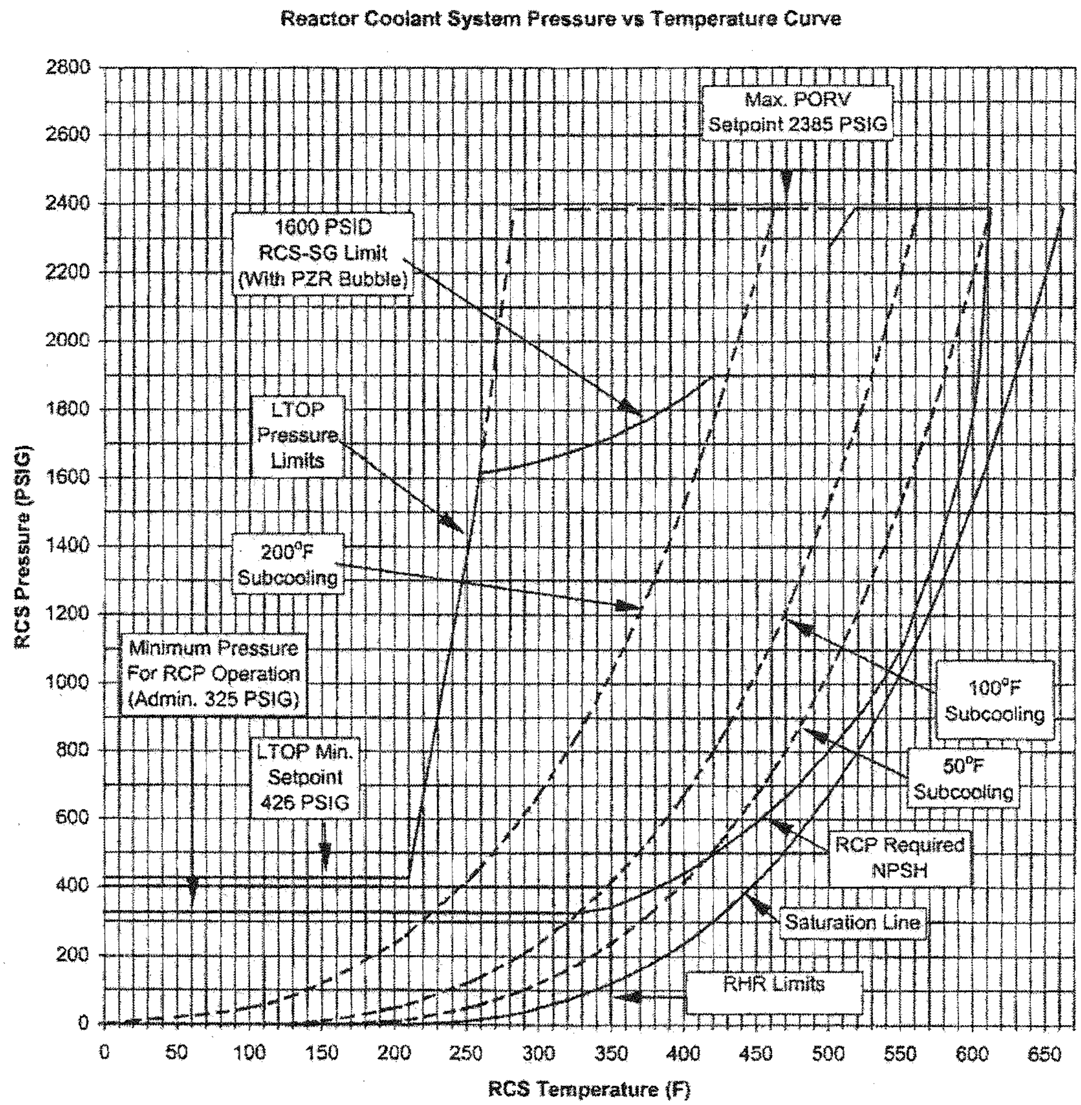
Question Cognitive Level: Memory or Fundamental Knowledge ______
Comprehension or Analysis X

10 CFR Part 55 Content:	55.41	41.10
	55.43	43.5

Comments:

Number	Title	Rev./Date
ES-0.3	**NATURAL CIRCULATION COOLDOWN WITH STEAM VOID IN VESSEL (WITH RVLIS)**	**27** **06/18/08**

FIGURE ES-0.3-1 PRESSURE TEMPERATURE LIMITS



Question 86, SRO 11

Examination Outline Cross-Reference:

		RO	SRO
	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	003A2.05	
	Importance Rating		2.8

003 = Reactor Coolant Pump

A2 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: A2.05 Effects of VCT pressure on RCP seal leakoff flows

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- Pressurizer Level Control is in Automatic
- "D4697, VCT PRESSURE LOW" is in alarm.
- The PSO reports VCT pressure is 10 psig and DECREASING SLOWLY.
- The Primary NSO discovers CS-PCV-8157, "VCT Pressure Control Valve" is failed OPEN.

Assuming Charging and Letdown flow have not changed, what actions must be taken in accordance with applicable plant procedures if the Crew is unable to correct this condition?

A. RCP #1 Seal Leakoff Flow will decrease. RCP #2 Seal Leakoff flow will decrease. If total seal leakoff flow decreases to less than 1.0 GPM and seal inlet temperature is stable THEN Shutdown the plant to MODE 3 within 8 hours and STOP the affected RCPs.

B. RCP #1 Seal Leakoff Flow will Increase. RCP #2 Seal Leakoff flow will decrease. IF total seal leakoff flow increases to greater than 8.0 GPM THEN TRIP the reactor and perform immediate actions of E-0, "Reactor Trip or Safety Injection", then STOP the affected RCPs.

C. RCP #1 Seal Leakoff Flow will Increase. RCP #2 Seal Leakoff flow will decrease. IF total seal leakoff flow increases to greater than 8.0 GPM THEN Shutdown the plant to MODE 3 within 8 hours while maintaining seal injection flow between 9.0 and 13.0 GPM, and WHEN plant is in MODE 3 then STOP the affected RCPs.

D. RCP #1 Seal Leakoff Flow will decrease. RCP #2 Seal Leakoff flow will decrease. If total seal leakoff flow decreases to less than 1.0 GPM and seal inlet temperature is stable THEN TRIP the reactor and perform immediate actions of E-0, "Reactor Trip or Safety Injection", STOP the affected RCPs, then CLOSE No. 1 seal leakoff valve.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. As VCT backpressure decreases RCP #1 Seal Leakoff Flow will Increase. With less backpressure to force the flow to RCP #2 Seal, the #2 seal leakoff flow will decrease. Because most of the total seal leakoff flow comes from the #1 seal, IF the VCT depressurization allows total seal leakoff flow to increase greater than 8.0 GPM THEN the procedure directs TRIP the reactor and perform immediate actions of E-0, "Reactor Trip or Safety Injection then STOP the affected RCPs.

Answer A is incorrect. RCP #1 Seal Leakoff Flow will increase. The procedure does direct that "If total seal leakoff flow decreases to less than 1.0 GPM and seal inlet temperature is stable THEN Shutdown the plant to MODE 3 within 8 hours then STOP the affected RCPs", so this can not be used to eliminate this distractor.

Answer C is incorrect. RCP #1 Seal Leakoff Flow will Increase, and RCP #2 Seal Leakoff flow will decrease but the action to "Shutdown the plant to MODE 3 within 8 hours while maintaining seal injection flow between 9.0 and 13.0 GPM, WHEN plant is in MODE 3 then STOP the affected RCPs" would only be performed if total leakoff flow was HIGH but between 6 and 8 GPM, not if seal leakoff flow was greater than 8 gpm (as stated in the distractor).

Answer D is incorrect. RCP #1 Seal Leakoff Flow will increase. Further the actions to "TRIP the reactor and perform immediate actions of E-0, "Reactor Trip or Safety Injection", then STOP the affected RCPs, WHEN RCP has coasted down then CLOSE associated No. 1 seal leakoff valve" are only performed if seal water inlet temperature in NOT stable or decreasing so this is also incorrect.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1201.01, "RCP Malfunction. Lesson L8021I, RCS Components.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8021I 28RO, L3020I 15SR, (As available)

Question Source: Bank #
Modified Bank # (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

Question Cognitive Level:	Memory or Fundamental Knowledge	_____
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 _____	
	55.43 43.5	

Comments:

Question 87, SRO 12

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	012 G 2.4.4	
	Importance Rating		4.7

012 = Reactor Protection
2.4 Emergency Procedures / Plan
2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

The following plant conditions exist:

- A plant shutdown is in progress.
- Power level is 30%
- The PSO notes that all UL-6 bistable lights for protection channel II have lit.
- The following VAS alarms have been received:

  D6006: "VITAL UPS 1B AC SUPPLY VOLTS LO"

  D6007: "VITAL UPS 1B AC OUTPUT VOLTS LO LO"

  D5801: "VITAL INST PANEL 1B POWER LOST"
- No other abnormal bistables were in the tripped condition prior to the event.
- All other plant parameters remain within normal operating limits during the event.

Which of the following describes the expected crew response to this condition?

A. The crew will enter E-0, "Reactor Trip of Safety Injection". Manual actuation of "B" train ESFAS components will be required.

B. The crew will enter OS1247.01, "Loss of a 120 VAC Vital Instrument Panel (PP1A, 1B, 1C, 1D)". Restore SG level by placing FRV control in manual.

C. The crew will enter ON1231.02, "Turbine Trip Below P-9". Stabilize Reactor Power and monitor secondary plant conditions during turbine coastdown.

D. The crew will enter E-0, "Reactor Trip of Safety Injection". RCS inventory control must be restored by placing Excess Letdown in service or Restoring Normal Letdown using OS1247.01, "Loss of a 120 VAC Vital Instrument Panel (PP1A, 1B, 1C, 1D)".

Proposed Answer: B

Explanation (Optional):

Answer B is correct. A Direct Reactor trip will only occur from loss of this vital Power panel if power level is below 8% (based on the loss of one Intermediate Range NI). The crew should take manual control of SG feedwater regulating valves to restore SG levels to normal.

Answer A is incorrect. The reactor will not trip at this power level. If an SI occurred the loss of PP-1B would require manual actuate of the associated train's equipment.

Answer C is incorrect. P-9 relates to reactor trip caused by turbine trip, but a turbine trip always occurs if a reactor trip occurs. No direct turbine trip will be caused by a loss of PP-1B.

Answer D is incorrect. A direct reactor trip will not occur at this power level. If a direct trip did occur the loss of PP-1B would cause a Normal letdown isolation so actions would be required for RCS inventory control.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Abnormal Operating Procedure OS1247.01, Loss of a 120VAC Vital Instrument Panel. PP-1A, 1B, 1C, or 1D.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8056I 17RO, L3022I 17SR, (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.2, 43.5

Comments:

Question 88, SRO 13

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	022A2.05	
	Importance Rating		3.5

022 = Containment Cooling
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the CCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.05 Major leak in CCS

Proposed Question:

The following plant conditions exist:

- The plant is operating at 18% power.
- The Main Turbine has just been phased on to the grid.
- The crew has entered OS1212.01, "PCCW System Malfunction", due to a RAPID DECREASE in train "B" PCCW head tank level.
- DM-V-17 has been opened to commence a head tank fill.
- Train "B" Head tank level is 32% and continues to DECREASE.

Based on these conditions, which of the following describes the required crew actions in accordance with OS1212.01?

A. Initiate a MANUAL Reactor trip and trip the "B" and "C" RCPs. Enter E-0, "Reactor Trip or Safety Injection" and perform Immediate Actions.

B. Initiate a MANUAL Reactor trip and enter E-0, "Reactor Trip or Safety Injection". When Immediate Actions are complete trip the "B" and "C" RCPs.

C. Isolate train "B" PCCW supply to WPB, SFP HX and RDMS Supply. Determine if the leak has been isolated as directed in OS1212.01, "PCCW System Malfunction".

D. Decrease Power to less than 10% within ten (10) minutes and STOP the B and C RCPs, Use OS1212.01, "PCCW System Malfunction" to isolate the leak and restore PCCW.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The Primary Component Cooling Water system is the heat sink for the Containment Structure Cooling fans at Seabrook. The Containment Structure Cooling fans are not safety related at Seabrook. In this question a leak in the PCCW system has lowered CC Head tank level below the containment automatic isolation setpoint. The RCPs cooled by the "B" PCCW loop (B and C) require shutdown. The procedure directs tripping the reactor prior to securing ANY RCPs. The affected RCPs are not secured until after the completion of E-0 immediate actions.

Answer A is incorrect. The procedure directs tripping the reactor and perform immediate actions of E-0 prior to securing ANY RCPs.

Answer C is incorrect. Head tank level is below the containment automatic isolation setpoint.

Answer D is incorrect. The abnormal always requires tripping the reactor and securing the RCPs if containment isolation valves are closed.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1212.01, "PCCW System Malfunction", step 5.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1445I 03RO, L3157I 03SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 89, SRO 14

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	026A2.04	
	Importance Rating		4.2

026 = Containment Spray
Ability to (a) predict the impacts of the following malfunctions or operations on the CSS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.04 Failure of spray pump

Proposed Question:

The following plant conditions exist:

- A LOCA has occurred
- CBS-P-9B tripped immediately after starting.
- The Crew has transitioned from E-0, "Reactor Trip or Safety Injection" to E-1, "Loss of Reactor or Secondary Coolant"
- When evaluating to determine if "Containment Spray Should Be Stopped" the PSO reports the following:
    - o CBS-P-9A Suction Pressure is 38 psig and SLOWLY DECREASING
    - o CBS-P-9A Discharge Pressure is 278 psig and SLOWLY DECREASING
    - o CBS-P-9A current is 54 amps and SLOWLY INCREASING
    - o Containment Pressure is 10 psig and SLOWLY DECREASING
    - o RWST level is 275,00 gallons and SLOWLY DECREASING

What is the status of CBS-P-9A and what actions are required?

A. The pump is operating normally. The crew should reset the P signal and the CBS signal then secure CBS-P-9A while remaining in E-1, "Loss of Reactor or Secondary Coolant".

B. The pump is at run-out flow. The crew should reset the P signal and the CBS signal, secure CBS-P-9A then transition to FR-Z.1, "Response to High Containment Pressure".

C. The pump is at shut-off head. The crew should reset the P signal and the CBS signal, secure CBS-P-9A then transition to FR-Z.1, "Response to High Containment Pressure".

D. The pump is operating normally. The crew should leave CBS-P-9A running and monitor Containment pressure while remaining in E-1, "Loss of Reactor or Secondary Coolant".

Proposed Answer: D

Explanation (Optional):

Answer D is correct. There is no direct indication available for CBS pump flow rate. Pump amps provides the best indirect indication of this parameter. The slowly increasing amps indicates that the pump is not cavitating and not at shutoff head. Oscillating amps would indicate cavitation. Abnormally low amps coupled with stable RWST level would indicate the pump is at shutoff head. Pump amps are increasing due to the dropping containment pressure. No procedure transition is required in this condition. The P signal will not be reset until Containment pressure is less than 4 psig while remaining in E-1, "Loss of Reactor or Secondary Coolant".

Answer A is incorrect. The pump is operating normally, however the P signal will not be reset until Containment pressure is less than 4 psig.

Answer B is incorrect. The slowly increasing amps, decreasing RWST level and lowering Containment pressure all indicate that the pump is not at runout flow. Amps are increasing due to increased pump flow as the containment pressure drops.

Answer C is incorrect. The slowly increasing amps, decreasing RWST level and lowering Containment pressure all indicate that the pump is not at shutoff head. Amps are increasing due to increased pump flow as the containment pressure drops.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
E-1, "Loss of Reactor or Secondary Coolant", step 7. FR-Z-1, "Response to High Containment Pressure", step 2

Proposed references to be provided to applicants during examination: None

Learning Objective: L8035I 10RO, L3041I 09SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question

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

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

10 CFR Part 55 Content:	55.41	41.5
	55.43	43.5

Comments:

Question 90, SRO 15

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	103 G 2.2.42	
	Importance Rating		4.6

103 = Containment
2.2 Equipment Control
2.2.42 Ability to recognize system parameters that are entry-level conditions for Technical Specifications.

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- The Containment Online Purge system is in service to control Containment Pressure during a prolonged summer heat wave.
- During performance of the T.S Surveillance shift logs the BOP operator notes that Containment Pressure is 16.5 psia as read on COP-PI-1787.
- Containment NR pressure indication SI-PI-937 indicates 1.9 psig.

What is the Technical Specification impact of this condition?

A. The two Containment Pressure instruments do not agree. The Technical Specification impact decision can be withheld until further instruments are checked to determine the required course of action.

B. Containment Pressure is above the maximum Technical Specification limits. The potential Containment Leakage may exceed the capability of the Containment Building Spray System to remove Iodine from the containment atmosphere. THROTTLE OPEN COP-V-7, "COP Exhaust Throttle Valve", to reduce containment pressure within 1 hour.

C. Containment Pressure is above the maximum Technical Specification limit pressure. The potential peak LOCA accident pressure may exceed the 52 psig design limit. THROTTLE OPEN COP-V-7, "COP Exhaust Throttle Valve", to reduce containment pressure within 1 hour.

D. Containment Pressure is below the minimum Technical Specification limits. The Containment Enclosure annulus may exceed its design negative pressure differential. THROTTLE CLOSE COP-V-7, "COP Exhaust Throttle Valve", to increase containment pressure within 1 hour.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. Containment Pressure is above the maximum Technical Specification limit pressure 16.2 psia. The bases for TS 3.6.1.4 states the potential peak LOCA accident pressure may exceed the 52 psig design limit. Throttling open COP-V-7, will reduce containment pressure.

Answer A is incorrect. The two Containment Pressure instruments do agree. One is given in PSIA and the other in PSIG.

Answer B is incorrect. Although the Containment Pressure is above the maximum Technical Specification limits, Iodine removal capability of the Containment Building Spray System is tied to the SAT tank and is independent of containment pressure.

Answer D is incorrect. Containment Pressure is above the maximum Technical Specification limit pressure 16.2 psia.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
T.S. and T.S bases for 3/4. 6.2.1, 3/4 . 6.1.4,

Proposed references to be provided to applicants during examination: None

Learning Objective: L3004I 11SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7, 41.10
55.43 43.2, 43.3

Comments:

Question 91, SRO 16

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	034K4.03	
	Importance Rating		3.3

034 = Fuel Handling Equipment
K4 Knowledge of design feature(s) and/or interlock(s) which provide for the following:
K4.03 Overload protection

Proposed Question:

The following plant conditions exist:

- The plant is in Mode 6
- Core reload is in progress
- The Work Control Supervisor reports that, due to a work control scheduling problem, the Refueling Machine Technical Requirements surveillance was not performed within 100 hours of the start of core reload.

What restrictions are placed on the core reload evolution and why?

A. Immediately suspend all movement of fuel assemblies within the reactor vessel. The FUNCTIONALITY of the 3900 lb. overload cutoff limit that protects the core internals and reactor vessel from excessive lifting force is unproven.

B. Immediately suspend all movement of fuel assemblies within the reactor vessel. Drive shaft movements may continue in the Refueling cavity providing a single-failure-proof NUREG-0554 compliant crane is used for any lift.

C. Movement of fuel assemblies within the reactor vessel can continue provided that a FUNCTIONAL Load Indicator with a minimum capacity of 2100 lbs is attached. This assures the hoist has sufficient load capacity to safely lift a fuel assembly.

D. Movement of fuel assemblies within the reactor vessel can continue provided that the 4000 lbs minimum hoist capacity portion of the Refueling machine surveillance had been proven FUNCTIONAL within 100 hours. This assures each hoist has sufficient load capacity to lift the combined weight of a drive shaft and a fuel assembly.

Proposed Answer: A

Explanation (Optional):

Answer A is correct. TR 26-3.9.6 requires immediately suspending all movement of drive shafts or fuel assemblies within the reactor vessel. The bases for the 3900 lb. overload cutoff limit is to protect the core internals and reactor vessel from excessive lifting force in the event they are inadvertently engaged during lifting operations.

Answer B is incorrect. The action given is correct (Immediately suspend all movement of fuel assemblies within the reactor vessel), but the bases for that action is incorrect. The control of lifts using a single-failure-proof NUREG-0554 compliant crane is associated with the bases for TR 27-3.9.7 Crane travel areas for Dry Cask Storage Loading operations.

Answer C is incorrect. Use of a Load Indicator with a minimum capacity of 2100 lbs in lieu of a load cutoff is only for use for latching and unlatching control rod drive shafts.

Answer D is incorrect. The Refueling machine requires BOTH a 4000 lbs minimum hoist capacity test AND a 3900 lb cutoff within 100 hours. The bases for this IS to assure sufficient load capacity to lift the combined weight of a drive shaft and a fuel assembly.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
TR26-3.9.6, Refueling Machine, bases additional information. TR27-3.9.7, Spent Fuel Pool Areas – Crane Travel, bases additional information.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8060I 08RO, L8060I 11SRO (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 43.7

Comments:

Question 92, SRO 17

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	041 G 2.4.9	
	Importance Rating		4.2

041 = Steam Dump/ Turbine Bypass Control
2.4 Emergency Procedures / Plan
2.4.9 Knowledge of low power/shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies

Proposed Question:

The following plant conditions exist:

- RCS temperature is 300 degrees F
- RCS pressure is 350 psig.
- The crew is performing a Plant cooldown using OS1000.15, "Refueling Outage Cooldown".
- When SG pressure reached less than 240 psig I&C bypassed the P-12 interlocks as directed.
- Train "A" RHR is aligned in the Shutdown cooling mode.
- Train "B" RHR is aligned in the ECCS mode.
- A tagging order error deenergizes bus 5. The Emergency Power Sequencer (EPS) re-powers bus 5 from the "A" Emergency Diesel Generator.
- The "A" RHR pump breaker trips and locks out when started by the EPS.
- All other equipment operates as designed.

What actions will the Unit Supervisor direct using OS1213.01, "Loss of RHR During Shutdown Cooling" to stabilize RCS temperature control?

A. Start Train "B" RHR and align to the shutdown cooling mode. Control RHR cooling flow as necessary to stabilize RCS temperature.

B. Steam dumps remained armed and reopened on power restoration. Control steam flow as necessary to stabilize RCS temperature.

C. Rearm the Steam dumps by taking both P-12 Interlock switches to INTLK BYPASS. Control steam flow as necessary to stabilize RCS temperature.

D. The Steam dump Cooldown Bank will remain armed and reopen on power restoration. Rearm the remaining Steam dumps by taking both P-12 Interlock switches to INTLK BYPASS. Control steam flow as necessary to stabilize RCS temperature.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. Prior to losing bus 5 ALL Steam dumps (vice just the cooldown bank) were available because the P-12 low temperature bistables were bypassed as directed in the Cooldown procedure. Steam dump capability is available from all banks because bus 5 was repowered and the P-12 bistables would remain bypassed. The steam dump controller output would need to be adjusted in order to compensate for the loss of cooling that was provided from the "A" RHR loop and to transition from the previous RCS cooldown to stabilizing RCS temperature.

Answer A is incorrect. IF the plant was in mode 5 or 6 then OS1213.01 would direct starting any other available RHR train and align it to shutdown cooling. When the plant is in mode 4 the procedure directs dumping steam to control RCS temperature.

Answer C is incorrect. The P-12 bistables would remain bypassed so it is not necessary to re-arm them.

Answer D is incorrect. All P-12 bistables would remain bypassed so it is not necessary to re-arm them.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OS1000.15, "Refueling Outage Cooldown, steps 4.2.7, 4.2.14. OS1213.01, "Loss of RHR During Shutdown Cooling", step 6.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1705I 02RO, L3159I 04SR (As available)

Question Source: Bank # _____
Modified Bank # _____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.5

Comments:

Question 93, SRO 18

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	075A2.02	
	Importance Rating		2.7

075 = Circulation Water
A2 Ability to (a) predict the impacts of the following malfunctions or operations on the circulating water system; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: A2.02 Loss of circulating water pumps

Proposed Question:

The following plant conditions exist:

- The plant is in Mode 1 at 100% power.
- CW screen differential level high alarms have been received.
- The Crew has entered ON1238.01, "Circulating Water System Malfunction", and is washing screens in fast speed.
- Condenser vacuum is 26.5" HG and slowly degrading.
- The Roving NSO reports CW screen differential levels as:
  - CW-SR-1A level is 56"
  - CW-SR-1B level is 54"
  - CW-SR-1C level is 60"
- The BOP operator reports that CW-V-11, "CW-P-39C Discharge", is indicating intermediate position.

Based on these conditions what procedural actions are required next?

A. Monitor CW-V-11 position. IF CW-V-11 reaches FULL CLOSED then direct the BOP operator to place the C" CW pump control switch in PTL.

B. Place the "C" CW pump control switch in PTL. Verify that the "C" CW pump is secured, then direct the BOP operator to CLOSE CW-V-11.

C. Direct the BOP operator to place the "C" CW pump control switch in STOP. Verify the "C" CW pump trips when CW-V-11 is less than 25% OPEN.

D. Direct the BOP operator to place the control switch for CW-V-11 to CLOSE. When CW-V-11 is less than 25% OPEN then direct the BOP operator to place the "C" CW pump control switch in STOP.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. When the pump control switch is placed in STOP then CW-V-11 will start to close. When the valve is less than 25% OPEN, the pump automatically trips, and the valve continues to full closed.

Answer A is incorrect. If no action is taken then CW-V-11 will continuously cycle open and closed as the differential pressure decreases when flow decreases. Placing the C" CW pump control switch in PTL will trip the pump immediately causing backflow through the pump.

Answer B is incorrect Placing the C" CW pump control switch in PTL will trip the pump immediately causing backflow through the pump.

Answer D is incorrect. The pump control scheme automatically secures the pump by placing the pump control switch in STOP which causes CW-V-11 to start close. When the valve is less than 25% OPEN, the pump automatically trips, and the valve continues to full closed.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ON1238.01, "Circulating Water System Malfunction", Caution prior to step 13 and step 13.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8053I 09RO, L3027I, 08SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.5
55.43 43.5

Comments:

Question 94, SRO 19

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	2.1.5	
	Importance Rating		3.9

2.1.5 Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc

Proposed Question:

The following plant conditions exist:

- It is 0400 on Wednesday September 15th.
- The plant is in Mode 4 for extended DG repairs.
- Only normal crew compliment is available.
    - 1 Shift Manager, 1 WCS, 1 US, 2 CROs, 5 NSOs, 1Fire Brigade Leader
- The Primary Side Control Room Operator becomes sick and must be taken to the hospital.
- No other licensed operators are on site.

What Actions, if any, are required?

A. No action is required. T.S. table 6.2-1 and the OPMM allow this position to remain vacant in this mode.

B. Action must be taken to obtain a replacement operator within 2 hours. The OPMM Overtime Distribution Call-In Instructions SHALL be used to fill the vacant position.

C. Action must be taken to obtain a replacement operator within 2 hours. The Shift manager may waive use of OPMM Overtime Distribution Call-In Instructions in this case.

D. No action is required provided that the Work Control Supervisor has an active license and assumes the RO position. Reassignment of Remote Safe Shutdown responsibilities to 2 qualified NSOs shall be documented in the Unit Journal.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The TS administrative section allows 2 hours to obtain a replacement operator in the event of an emergency. The OPMM allows the Shift manager to waive use of OPMM figure 2-1-2 "Overtime Call Instructions" in this case.

Answer A is incorrect. T.S. table 6.2-1 and OPMM figure 2-1-1, "Minimum Shift Composition" identifies that this position must be filled in mode 4.

Answer B is incorrect. The OPMM states that the "Overtime Call Instructions" in OPMM figure 2-1-2 may be waived to fill the TS required vacant position.

Answer D is incorrect. 4 qualified NSOs are required to be assigned as fire brigade members. The 5th NSO and the WCS make up the 2 required Remote Safe Shutdown operators. The WCS is qualified, but not available to assume the RO responsibilities.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
OPMM, Chapter 2, section 1.5.6, Figure 2-1-1, Figure 2-1-2. T.S. Section 6, Section 6.2 Station Staff, Table 6.2-1

Proposed references to be provided to applicants during examination: None

Learning Objective: L1505I 02RO & 03RO, L3002I 10SR (As available)

Question Source: Bank #
Modified Bank # 26762 (Note changes or attach parent)
New

The following plant conditions exist:

- The plant is in Mode 4.
- The Primary Side Control Room Operator becomes sick and must be taken to the hospital.
- There is only one other qualified RO/SRO on site who is standing watch in the control room currently as the secondary control room operator.
- There are 3 hours left until shift turnover.

What Action is required?

A. The affected RO must not be allowed to leave the site until a relief operator arrives.

B. Immediate action must be taken to obtain a replacement operator within 4 hours.

C. Immediate action must be taken to obtain a replacement operator within 2 hours.

D. Shift turnover will occur before any action is required. Action should be made to find a replacement but is not required.

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.10
	55.43	43.2, 43.5

Comments:

Question 95, SRO 20

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.1.45	
	Importance Rating		4.3

2.1.45 Ability to identify and interpret diverse indications to validate the response of another indication.

Proposed Question:

The following plant conditions exist:

- A LOCA has occurred.
- SI alignment has been verified.
- Current plant instrumentation readings are as follows:
  - Train "A" RCS Subcooling indicates 35 degs F.
  - Train "B" RCS Subcooling indicates 0 degs F.
  - Pressurizer pressure indicates 1600 psig.
  - RCS Wide Range pressure indicates 1470 psig.
  - Hot Channel Core Exit thermocouple indicates 560 degs F.

What is actual RCS subcooling and what action should be taken based on these indications?

A. Actual RCS subcooling is 34 degs F. For Small Break LOCA conditions the primary concern is protection of the RCP for later use.

B. Actual RCS subcooling is 34 degs F. All Reactor Coolant pumps are secured following a small break LOCA to prevent loss of excessive inventory when the RCPs are tripped.

C. Actual RCS subcooling is 46 degs F. All Reactor Coolant pumps should remain running if possible to provide normal Pressurizer Spray flow and forced RCS flow.

D. Actual RCS subcooling is 46 degs F. All Reactor Coolant pumps should remain running if possible because it is desirable to minimize operator actions such as tripping the RCPs then restarting them later.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. RCS subcooling is 34 degees F. This is based on saturation temperature for an RCS pressure of 1470 psig (Wide range pressure) (594.8 degrees), as compared to the Core Exit Thermocouple temperature of 560 degrees. The WOG Generic states that all Reactor Coolant pumps should be secured following a small break LOCA to prevent loss of excessive inventory when the RCPs are tripped.

Answer A is incorrect. While the subcooling number is correct, the reason provided why the RCPs are secured is incorrect. For Small Break LOCA conditions the primary concern is prevention of mass loss when RCP eventually trips, protection of equipment is a secondary concern.

Answer C is incorrect. The given RCS subcooling is 46 degrees F is based on the Pressurizer pressure indication of 1600 psig which is the bottom "peg" of indicated pressure for that instrument. Actual RCS pressure is below the indication range for the PZR pressure instrument. The WOG generic issue volume does state that Reactor Coolant pumps should not be tripped early so that they can remain running if possible to provide normal Pressurizer Spray flow and forced RCS flow.

Answer D is incorrect. The given RCS subcooling is 46 degrees F is based on the Pressurizer pressure indication of 1600 psig which is the bottom "peg" of indicated pressure for that instrument. Actual RCS pressure is below the indication range for the PZR pressure instrument. The WOG generic issue volume does state that all Reactor Coolant pumps should remain running if possible because it is desirable to minimize operator actions such as tripping the RCPs then restarting them later.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
Steam Tables, and Westinghouse Owners group background documents Executive Summary, RCP trip criteria.

Proposed references to be provided to applicants during examination: Steam Tables

Learning Objective: L8058I 11RO, L3040I 10SR (As available)

Question Source: Bank # ____
Modified Bank # 27325 (Note changes or attach parent)
New ____

27325 (Modified)

The following plant conditions exist:

- A LOCA has occurred.
- RCS Wide Range pressure has stabilized at 1559 psig.
- Highest Average Quadrant CETC temperature is 570 degs F
- Highest CETC temp is 590 degs F

- Highest Wide Range Thot indicates 545 degs F

What is the expected Subcooling Monitor reading and the condition of the RCS?

A. -40 degs F, superheated

B. +20 degs F, subcooled

C. +55 degs F, superheated

D. +40 degs F, subcooled.

Answer = D

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.7
	55.43	43.5

Comments:.

Question 96, SRO 21

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.21	
	Importance Rating		4.1

2.2.21 Knowledge of pre- and post-maintenance operability requirements

Proposed Question:

The following plant conditions exist:

- The Plant is at 100% power.
- The Primary NSO has reported water appears to be dripping on RH-V-70, RHR TRAIN A COMM SUP TO HL RECIRC.
- The water is dripping on a valve limit switch then along the valve stem.
- A catch basin has been set up to collect the leakage.
- There appears to be evidence of old electrical burn marks on the valve motor.
- The Shift Manager is attempting to determine Valve Operability.

Which of the following courses of action would be an acceptable way of determining valve operability?

A. Clean and lubricate the Valve Stem and the Limit switch only. Successfully stroke and time the valve while the NSO observes the valve to document operability.

B. Have the NSO stand clear and stroke the valve OPEN and CLOSE from the Main Control Board. IF the valve strokes successfully, THEN stroke and time the valve while the NSO observes the valve to document operability.

C. OPEN and danger tag the valve breaker. Remove the case and inspect the Motor. The EARLIEST time the valve can be returned to OPERABLE would be after it is successfully stroked and timed while the NSO observes the valve to document operability.

D. Have the NSO de-clutch and locally OPEN and CLOSE the valve. The valve remains OPERABLE provided that the NSO remains at the valve while it is de-clutched. IF the valve strokes smoothly, THEN stroke and time the valve while the NSO observes the valve to document operability.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. Evidence of electrical burn marks requires inspection of the valve motor. Stroking and timing the valve will be required to demonstrate operability once maintenance is complete.

Answer A is incorrect. Cleaning and lubricating constitutes preconditioning and does not address the burn marks.

Answer B is incorrect. Exercising the valve prior to as found testing constitutes preconditioning and does not address the burn marks.

Answer D is incorrect. Manually stroking the valve prior to as found testing constitutes preconditioning and does not address the burn marks.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
WM 8.0, section 4.2, Preconditioning of Equipment.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1514I 03SR (As available)

Question Source: Bank # ____
Modified Bank # ____ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.10
55.43 43.2

Comments:

Question 97, SRO 22

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	2.2.37	
	Importance Rating		4.6

2.2.37 Ability to determine operability and/or availability of safety related equipment.

Proposed Question:

The following plant conditions exist:

- The plant is at 100% power.
- At 0800 the Engine Driven Lube Oil Pump for the "A" Emergency Diesel Generator (EDG) failed as the EDG was being secured from a scheduled DG surveillance.
- The Auxiliary Motor Driven Lube Oil pump started as designed and supplied engine lube oil during the completion of the engine shutdown with no Lube Oil leakage noted.
- A replacement pump will arrive by 1200 tomorrow.

What action is required in accordance with Technical Specifications?

A. The "A" DG is FUNCTIONAL and OPERABLE. The applicable Technical Specification is satisfied.

B. The "A" DG became INOPERABLE, but FUNCTIONAL when the pump failed. No surveillance has to be performed on the "B" Emergency Diesel Generator for up to 72 hours provided that SEPS is FUNCTIONAL.

C. The "A" DG became INOPERABLE, but FUNCTIONAL when the pump failed. A surveillance run must be performed on the "B" Emergency Diesel Generator by 0800 tomorrow or be in at least HOT STANDBY within the next 6 hours.

D. The "A" DG became INOPERABLE and NONFUNCTIONAL when the pump failed. A surveillance run must be performed on the "B" Emergency Diesel Generator by 0800 tomorrow or be in at least HOT STANDBY within the next 6 hours.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. The "A" DG became INOPERABLE when the pump failed because the engine driven lube oil pump is a required subsystem required for operability. The auxiliary pump supports the EDG remaining FUNCTIONAL. A surveillance run on the "B" Emergency Diesel Generator within 24 hours to prove it does not have a common mode failure.

Answer A is incorrect. The "A" DG became INOPERABLE when the pump failed because the engine driven lube oil pump is a required subsystem required for operability.

Answer B is incorrect. SEPS allows for extension of planned outage time, and is not tied to the 24 requirement to test the other DG for common mode failure considerations.

Answer D is incorrect. The auxiliary pump supports the EDG remaining FUNCTIONAL.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
T.S 3.8.1.1, action b.

Proposed references to be provided to applicants during examination: None

Learning Objective: L8020I 24RO, L3002I 10SR (As available)

Question Source: Bank # ______
Modified Bank # ______ (Note changes or attach parent)
New X

Question History:
Last NRC Exam N/A new question
Exam Bank History N/A new question
(Optional: Questions validated at the facility since 10/95 will generally undergo less rigorous review by the NC, failure to provide the information will necessitate a detailed review of every question.)

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

10 CFR Part 55 Content: 55.41 41.7
55.43 43.2

Comments:

Question 98, SRO 23

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #	____	3
	Group #	____	3
	K/A #	2.3.13	
	Importance Rating	____	3.8

2.3.13 Knowledge of radiological safety procedures pertaining to licensed operator duties, such as response to radiation monitor alarms, containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question:

The following radiological conditions exist in the Letdown degassifier valve room:

- General Dose rates range from 25 – 45 mrem/hr
- Measurements taken on pipes and valves are:
    - Point 1 is 100 mrem/hr at 30 cm
    - Point 2 is 500 mrem/hr at 30 cm.
    - Point 3 is 1100 mrem/hr at 30 cm
- The room needs to be accessed by an NSO.
- No RWP has been authorized for this area.
- No HP supervisor can be reached.

Based on these conditions what should be the radiological posting for this room, and who can authorize a RWP to allow entry by the NSO?

A. High Radiation Area, Only a HP Supervisor

B. High Radiation Area, the Station Director and the Shift Manager.

C. Technical Specification Locked High Radiation Area, the Shift Manager if no HP supervisor can be reached.

D. Technical Specification Locked high Radiation Area, only the Station Director if no HP supervisor can be reached.

Proposed Answer: C

Explanation (Optional):

Answer C is correct. Radiation levels greater than 1000 mrem/hr make this a Technical Specification Locked High Radiation Area. The Shift Manager can authorize entry if no HP supervisor can be reached.

Answer A is incorrect. Radiation levels greater than 1000 mrem/hr make this a Technical Specification Locked High Radiation Area.

Answer B is incorrect. Radiation levels greater than 1000 mrem/hr make this a Technical Specification Locked High Radiation Area..

Answer D is incorrect. Radiation levels greater than 1000 mrem/hr make this a Technical Specification Locked High Radiation Area. The Shift Manager can authorize entry if no HP supervisor can be reached, in lieu of the Station Director.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
RP 9.1, section 4.1, page 8, RP chapter 1, 1.2.5, page 1-1.3

Proposed references to be provided to applicants during examination: None

Learning Objective: L1229I 03RO (As available)

Question Source: Bank #
Modified Bank # 26978 (Note changes or attach parent)
New

Original 26978

The following radiological conditions exist for a room in the plant:

- General Dose rates range from 25 – 45 mrem/hr
- Measurements taken on pipes and valves are:
    - Point 1 is 100 mrem/hr at 30 cm
    - Point 2 is 500 mrem/hr at 30 cm.
    - Point 3 is 1100 mrem/hr at 30 cm
- The room is accessible to station personnel.

Based on these conditions what should be the radiological posting required for this room, and who can authorize an individual to exceed Federal Annual TEDE limits while working in this room during a non-emergency situation?

A. High Radiation Area, Station Director.

B. Technical Specification Locked High Radiation Area, Station Director.

C. High Radiation Area, Nobody can authorize exceeding the Federal Limits.

D. Technical Specification Locked high Radiation Area, Nobody can Authorize exceeding the Federal Limits.

Answer D

Question History:

Last NRC Exam	Original question never used on a NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.12
	55.43	43.4

Comments:

Question 99, SRO 24

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.4	
	Importance Rating		4.7

2.4.4 Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question:

The following plant conditions exist:

- A Plant shutdown is in progress
- Reactor Power is 18%
- The Main Generator is Synched to the Grid
- Steam Dumps are in Steam Pressure mode in Automatic
- Level in "B" S/G swells to 91%
- Level in the "B" SG is presently at 88% and slowly decreasing
- Tavg is 561 degs and slowly DECREASING.

Which of the following describes the crew response to stabilize the plant?

A. A Manual Turbine Trip and Reactor Trip is required. Enter E-0, "Reactor Trip or Safety Injection". Throttle EFW flow to restore SG level.

B. An Automatic Turbine Trip and Reactor Trip occurred. Enter E-0, "Reactor Trip or Safety Injection". Throttle EFW flow to restore SG level.

C. An Automatic Turbine Trip occurred. Enter ON1231.02, "Turbine Trip Below P-9". No Reactor Trip occurred or is necessary. RESET FWI and manually control SG level between 45% and 55%, as necessary.

D. An Automatic Turbine Trip occurred. Enter ON1231.02, "Turbine Trip Below P-9". A Reactor Trip is required because SG level control is unrecoverable. Enter E-0, "Reactor Trip or Safety Injection". Throttle EFW flow to restore SG level.

Proposed Answer: D

Explanation (Optional):

Answer D is correct. An Automatic Turbine Trip occurred due to P-14 (Hi SG water level). After entering ON1231.02, "Turbine Trip Below P-9" the crew will evaluate SG level trending to 50% in step 3b. The RNO will direct a Reactor Trip because SG levels will continue to decrease after the Main Feeds Tripped on High SG level. This requires entry into E-0, "Reactor Trip or Safety Injection". Throttle EFW flow to restore SG level.

Answer A is incorrect. An automatic Turbine Trip has already occurred.

Answer B is incorrect. An Automatic Reactor Trip did not occur because power level is below the P-9 setpoint.

Answer C is incorrect. Resetting the FWI will not matter because the main feed pumps tripped on high SG level. A Manual Reactor Trip will be required.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ON1231.02, "Turbine Trip Below P-9, Step 3. FWI circuitry, Main Feed pump Trip Circuitry, Reactor Trip /P-9 Interlock.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1183I 06RO, L1183I 07RO, L3022I 08SR (As available)

Question Source: Bank # _____
Modified Bank # 25126 (Note changes or attach parent)
New _____

Original 25126

The following plant conditions exist:

- A Plant shutdown is in progress
- Reactor Power is 12%
- Steam Dumps are in Steam Pressure mode in Automatic
- Level in "B" S/G swells to 88%, due to PT-507 failing high
- Tavg stabilizes at 554 degs F after the Steam Dumps are shut.
- Level in the "B" SG is presently at 83% and slowly decreasing.

Which of the following describes the response of the Feedwater Regulating Bypass Valves, bases on the above conditions?

A. The Feedwater regulating bypass valves will remain closed.

B. The Feedwater regulating bypass valves will return to controller demand positions

C. The Feedwater regulating bypass valves will remain closed until the Operator momentarily places both Feedwater Isolation Reset switches in the RESET position.

D. The Feedwater regulating bypass valves will remain closed until the Operator momemtarily places the Reactor Trip Reset control Switches in the RESET position.

Answer = B

Question History:

Last NRC Exam	Never used on NRC Exam
Exam Bank History	Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.10
	55.43	43.5

Comments:

Question 100, SRO 25

Examination Outline Cross-Reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	2.4.29	
	Importance Rating		4.4

Comments: 2.4.29 Knowledge of the emergency plan

Proposed Question:

The following plant conditions exist:

- A SITE AREA EMERGENCY was declared 37 minutes ago.
- The Emergency Response plan facilities have NOT been activated yet.
- The on shift Work Control Supervisor has made the Notification to the States and NRC.
- Conditions have stabilized and the event no longer meets the Emergency Action Level criteria.

Who is responsible for termination of the classification?

A. ONLY the Response Manager

B. Response Manager or Site Emergency Director.

C. Response Manager or Short Term Emergency Director.

D. Site Emergency Director or Short Term Emergency Director.

Proposed Answer: B

Explanation (Optional):

Answer B is correct. The Response Manager or Site Emergency Director can terminate the radiological event.

Answer A is incorrect. The Response Manager or Site Emergency Director can terminate the radiological event.

Answer C is incorrect. The Short Term Emergency Director does not have the authority to terminate a radiological event.

Answer D is incorrect. The Short Term Emergency Director does not have the authority to terminate a radiological event.

Technical Reference(s):
(Attach if not previously provided) (including version/revision number)
ER 1.1, section 1.1. , ER 1.2, SAE checklist for STED. ER 1.2 precaution 3.5.

Proposed references to be provided to applicants during examination: None

Learning Objective: L1509I 09SR (As available)

Question Source: Bank #
Modified Bank # 28038 (Note changes or attach parent)
New

TEB 28038

The following plant conditions exist:

- A SITE AREA EMERGENCY was declared 37 minutes ago.
- Notification has been made to the States and NRC.
- Conditions have stabilized and the event is terminated.

Who is responsible for termination of the classification?

A. Licensing Coordinator

B. Response Manager

C. Short Term Emergency Director

D. Emergency Operations Manager.

Answer = B

Question History:
Last NRC Exam Never used on NRC Exam

Exam Bank History Modified from bank

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

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

10 CFR Part 55 Content:	55.41	41.10
	55.43	43.5

Comments: