Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	261000 K1.07	
	Importance Rating	3.1	

Knowledge of the physical connections and/or cause- effect relationships between STANDBY GAS TREATMENT SYSTEM and the following: Elevated release stack

Proposed Question: RO Question # 1

Given the following:

- Accident conditions have resulted in a requirement to vent the primary containment.
- Torus pressure is 10 PSIG up slow.
- Torus water level is 15 feet and steady.
- Two reactor building blowout panels have ruptured.
- All other plant equipment is available.

Which vent path will ensure a release path through the primary vent stack?

Use OE 3107, Appendix HH…

A. Section #2, 18” air purge RTF’s to stack.

B. Section #6, 4” sprays to waste collector to RW to stack.

C. Section #8, 3” vent to SBGT to stack.

D. Section #10, 3" vent to RBHVAC to stack.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Failure at containment pressure >2 psig can result in an unmonitored release from the reactor building roof.

B. Incorrect. Containment pressure must be greater than 20 psig to use this path.

C. Correct.

D. Incorrect. Failure at containment pressure >2 psig can result in an unmonitored release from the reactor building roof.

Technical Reference(s): OE 3107, App HH, Table 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: OE-3107, Appendix HH

Learning Objective: LOT-00-261 k1.07

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 4

55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K1.06	
	Importance Rating	3.7	

Knowledge of the physical connections and/or cause- effect relationships between HIGH PRESSURE COOLANT INJECTION SYSTEM and the following: Suppression chamber: BWR-2,3,4

Proposed Question: RO Question # 2

The plant is operating with the following conditions:

- HPCI is injecting water from the CST to the RPV.
- HPCI torus suction valves HPCI-57 and HPCI-58 control switches are in Auto.
- CST low level alarm (3-S-4) comes in.

Which one of the following describes the expected response of the HPCI system?

The HPCI suction from CST 23-17 will receive a close signal as soon as …

A. both the HPCI-57 and the HPCI-58 valves reach full open.

B. both the HPCI-57 and the HPCI-58 valves come off their closed seats.

C. either the HPCI-57 or the HPCI-58 valve reaches full open.

D. HPCI-57 and HPCI-58 valves receive an open signal.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - This could potentially compromise HPCI availability when it is needed for injection if the system waited until a full open position.
B. Correct Response: To ensure continued HPCI suction path(s) the torus suction valves will open before closing off the CST suction path.
C. Incorrect - This could potentially compromise HPCI availability when it is needed for

injection if the system waited until a full open position.

D. Incorrect - The system needs a positive response from the suction valves making this a plausible distractor.

Technical Reference(s): LOT-00-206 (p, 34, 38; Rev 40) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # VY 3287
X
Modified Bank #

New

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 K2.01	
	Importance Rating	2.9	

Knowledge of electrical power supplies to the following: CCW pumps

Proposed Question: RO Question # 3

The plant was operating at 100% with the following conditions:

- ‘A’ RBCCW pump was running and the ‘B’ RBCCW pump is in AUTO.

  Then, the following event occurred:
- A LOCA occurred and the reactor scrammed on Hi drywell pressure.
- An LNP occurred due to a fault in the Start-Up transformers.
- Both Diesel Generators started and loaded simultaneously.

Based on these events, which ONE of the following is the expected response of the RBCCW pumps?

A. The 'A' pump will auto start 60 seconds after Bus 4 is re-energized.
The 'B' pump will auto start only if the 'A' pump fails to start.

B. The 'A' and 'B' pumps will auto start 60 seconds after Buses 3 and 4 are re-energized.

C The 'B' pump will auto start 60 seconds after Bus 3 is re-energized.
The 'A' pump will auto start only if the 'B' pump fails to start.

D. Neither pump will auto start and one or both pumps must be manually started.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The 'B' pump will auto start 60 seconds after Bus 3 and Bus 8 are re-energized. Because the buses are re-energized at exactly the same time, both pumps

should get an auto start signal because the low RBCCW header pressure signal is seen by both pumps.

B. Correct - Buses 9 and 8 power RBCCW pumps 'A' and 'B' respectively. Buses 9 and 8 are powered from Buses 4 and 3 respectively. Buses 9 and 8 remain connected to Buses 4 and 3 following LNP load shedding. RBCCW pumps are sequenced on 60 seconds following the EDG repowering the buses even when a LOCA signal is present.

C. Incorrect - The 'A' pump will auto start 60 seconds after Bus 4 and Bus 9 are re-energized. Because the buses are re-energized at exactly the same time, both pumps should get an auto start signal because the low RBCCW header pressure signal is seen by both pumps.

D. Incorrect - Buses 9 and 8 power RBCCW pumps 'A' and 'B' respectively. Buses 9 and 8 are powered from Buses 4 and 3 respectively. Buses 9 and 8 remain connected to Buses 4 and 3 following LNP load shedding. RBCCW pumps are sequenced on 60 seconds following the EDG repowering the buses even when a LOCA signal is present.

Technical Reference(s): OP 2182 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: LOT-00-208 (needed) (As available)

Question Source: Bank # (VY Bank)

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Vermont Yankee Bank

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 K2.02	
	Importance Rating	2.7	

Knowledge of electrical power supplies to the following: Analog trip system logic cabinets

Proposed Question: RO Question # 4

Which ONE of the following is supplied power by the Reactor Protection System?

A. 'D' main steam line radiation monitor

B. ADS System Logic Power

C. Core Spray System Logic Power

D. SLC Squib Valves

Proposed Answer: A

Explanation (Optional):

A. Correct. Rad monitors are powered from RPS

B. Incorrect. ADS logic cabinet is powered from DC-2C and DC-1C

C. Incorrect. CS logic is powered from DC-2C and DC-1C

D. Incorrect. SLC Squib Valves are powered from MCC-9B and MCC-8B

Technical Reference(s): LOT-00-212 pg. 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-212 EO K2.02 (As available)

Question Source: Bank # WTSI 2164

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2002 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 K3.02	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) will have on following: Recirculation pump speed: Plant-Specific
Proposed Question: RO Question # 5

Given the following:

- The plant is operating at power.
- Vital AC is lost.

Which ONE of the following describes how the Recirculation pumps will be affected prior to taking any operator action?

A. Recirc Pumps will trip.

B. Recirc Pumps will runback to minimum speed.

C. Automatic and remote manual Recirc Pump speed control will be lost.

D. Automatic Recirc Pump speed control will be lost.
Remote manual Recirc Pump speed control remains available.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect:** This is incorrect because the Scoop Tube Positioner will Lockup as a result of a loss of power, and must be reset before speed of the pump can be changed. This is plausible because there are plant conditions that will result in a recirc pump trip

B. **Incorrect:** This is incorrect because the Scoop Tube Positioner will Lockup as a result of a loss of power, and must be reset before speed of the pump can be changed. This is plausible because according to LOT-00-202 (p48; Rev 43), there are certain plant conditions that will cause the Recirc Pumps to runback to minimum speed (i.e. Feedwater flow < 18% for 15 seconds), and the operator may incorrectly believe that loss of power is one of them.

C. **Correct:** According to Lesson Plan LOT-04-262 (p23; Rev 20) Vital AC provides power to the A and B Recirc MG Scoop Tube Positioners; and on a loss of power Recirc MG A and B Scoop Tube Positioners lock up. Thus the operator will not have the ability to control Recirc MG speed and thus will not be able to adjust recirculation flow. The speed will remain as it was just prior to the loss of Vital AC. According to OP2144 (p36; Rev 45), both the A and the B Recirc Pump Scoop Tube Positioner is powered from Vital AC VAC-B Subpanel.

D. **Incorrect:** This is incorrect because both automatic and manual speed control are affected in accordance with ON-3168. With no vital AC available, all speed control is lost.

Technical Reference(s): Lesson Plan LOT-04-262 (p23; Rev 20) (Attach if not previously provided)
ON 3168

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-04-262 Objective EOK3.02 (As available)

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K3.21	
	Importance Rating	2.6	

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on following: Traversing in-core probe system

Proposed Question: RO Question # 6

The following conditions exist:

An OD-1 Full Tip set was in progress IAW OP 2425, Core power distribution calculation, utilizing the TIP system.
TIP machine #1 was in use.
The TIP ball valve and purge valve both indicated open.

Subsequently, due to a FWLC transient RPV level reached 120 inches.
A reactor scram occurred.
Five minutes later the TIP ball valve and purge valves continue to indicate open.
The CRS orders the TIP shear valve to be fired.

Which ONE of the following describes the indications that will be observed as a result of the shear valve being fired?

A. At CRP 9-13, ALL TIP shear valves indicate closed and the TIP purge valve is closed.

B. At CRP 9-13, ONE TIP shear valve indicates closed and the TIP purge valve is open.

C. At CRP 9-3 the TIP ball valve will indicate closed when ANY ball valve is closed or the shear valve is fired.

D. At CRP 9-3 the TIP ball valve will indicate closed when ALL ball valves are closed or the shear valve is fired.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. With three TIP machines and one in use, only one of the three shear valves would be fired. None of the other ball valves would be open with one TIP machine in use. The TIP purge valve was indicated in the stem to be open post isolation signal and it would not close due to shear valve operation.

B. Correct. One TIP shear valve has been fired and the purge valve would still indicate open for the reason stated in distractor 'A'.

C. Incorrect. It is true that the TIP ball valve indicates closed when ALL ball valves are closed, not any. It indicates open when any ball valve is open. The shear valve does not affect purge valve operation.

D. Incorrect. It is true that the TIP ball valve indicates closed when ALL ball valves are closed, not any. It indicates open when any ball valve is open. The shear valve does not affect purge valve operation.

Technical Reference(s): OP 2115, Primary containment (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-05-215 K6.04 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New XX

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000	K4.06
	Importance Rating	3.5	

Knowledge of REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) design feature(s) and/or interlocks which provide for the following: Manual initiation

Proposed Question: RO Question # 7

Given the following conditions:

- A transient has occurred requiring Control Room evacuation.
- All required Immediate Actions per OPOP-ALTSD-3126 "Shutdown Using Alternate Shutdown Methods" have been completed.
- RCIC is being operated for RPV level control from the RCIC Alternate Shutdown Panel (ASP).
- The RCIC turbine stopped.

WHICH ONE of the following describes what occurred to the RCIC System, and the current status of the system?

A RCIC…

A. turbine trip setpoint has been exceeded that can be locally reset, allowing RCIC to be restarted from the ASP.

B. turbine trip setpoint has been exceeded that cannot be reset locally, preventing RCIC from being restarted from the ASP.

C. system isolation setpoint has been exceeded, with RCIC restart possible once the isolation signal is reset from the ASP.

D. system isolation setpoint has been exceeded, and RCIC is no longer available for reactor water level control from the ASP.

Proposed Answer: A

Explanation (Optional):

A. Correct Response

B. Incorrect - all trips and isolations are bypassed when in EMERGENCY except turbine overspeed

C. Incorrect - all isolations are bypassed when in EMERGENCY

D. Incorrect - all isolations are bypassed when in EMERGENCY

Technical Reference(s): OPOP-ALTSD-3126 (Attachment 3 Step 7 CAUTION, pg. 2) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-612 Obj A (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 K4.08	
	Importance Rating	2.7	

Knowledge of AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM design feature(s) and/or interlocks which provide for the following: Sampling of overall core power in each APRM (accomplished through LPRM assignments and symmetrical rod patterns)
Proposed Question: RO Question # 8

Which ONE of the following identifies how APRM channels sample core detectors?

Each channel monitors…

A. the detectors in a separate horizontal plane and due to control rod symmetry is representative of average core power.

B. the detectors in a separate vertical sector and due to control rod symmetry is representative of average core power.

C. a group of detectors chosen in an orderly pattern across the core and includes detectors in vertical and horizontal planes.

D. a group of detectors chosen in an orderly pattern within separate vertical sectors.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect:** This is incorrect because all APRM channels have full core coverage (At least one Detector from every LPRM String). This is plausible because the operator may incorrectly believe that no APRM channels have full core coverage, but their assignment is sectionalized.
B. **Incorrect:** This is incorrect because all APRM channels have full core coverage (At least one Detector from every LPRM String). This is plausible because the operator may incorrectly believe that since APRM Channels B and E are the only APRMs that use

unshared LPRMs, only these will have full core coverage.

C. **Incorrect:** This is incorrect because all APRM channels have full core coverage (At least one Detector from every LPRM String). This is plausible because the operator may incorrectly believe that since APRM Channels A and D, and C and F are the only APRMs that use shared LPRMs, only these will have full core coverage.

D. **Correct:** According to LOT-03-215 (p12 and 24; Rev 15), the combination of the symmetrical distribution and the vertical distribution of detectors provides an accurate indication throughout the core because power is distributed evenly. The method of LPRM assignment is in an orderly pattern across core, selected so that each APRM will give good core average power signal, and are symmetric on diagonals through LPRM string. The assignments include APRM Channels A, C, and E and LPRM Group A, in one set of diagonals, and APRM Channels B, D, and F and LPRM Group B in the alternate set of diagonals. According to LOT-03-215 (p23 and 28; Rev 15), there are six channels (A-F) that receive signals from 80 in-core LPRM detectors. Each APRM receives the output of 20 detectors. Channels A and D share 20 detectors. Channels C and F share 20 detectors, and Channels B and E have 20 unshared detectors each. There are 20 strings of detectors with four levels of detectors on each string. According to Figure 1 of OP2456 (p1 of 1; Rev 6), shows the assignment of each LPRM. Both B and E use 20 un-shared detectors, one at a distinct level from every string. Additionally, Channels A, C, D and F have 20 shared detectors, also one at a distinct level from every string. Consequently, all APRM channels have full core coverage.

Technical Reference(s): LO-03-215 (p12, 23-24 and 28; Rev 15)
Figure 1 of OP2456 (p1 of 1; Rev 6)
(Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: None

Question Source: LOT-03-215 Objective K4.08 (As available)

Bank #

Modified Bank #

Question History: New X

Question Cognitive Level: Last NRC Exam: NA

Memory or Fundamental Knowledge X

10 CFR Part 55 Content:	Comprehension or Analysis	
Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.	55.41	7
Comments:	55.43	

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000	K5.13
	Importance Rating	2.9	

Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM: Filters

Proposed Question: RO Question # 9

Given the following:

- The crew is performing actions of OPON-3146-01, Loss of Instrument/Scram Air Header pressure.
- Alarm, 6-D-1, Lo Air HDR pressure is received.
- Instrument air pressures indicates 85 psig and lowering slowly on ‘A’ header and 100 PSIG and steady on the ‘B’ header.
- Air compressors ‘A’ and ‘C’ are running in Lead with normal indications.
- ‘B’ and ‘D’ air compressors are in Lag.
- Scram air header pressure indicates approximately 70 psig and stable.
- ‘A’ and ‘B’ air dryer local outlet pressures indicate approximately 100 psig and stable.

Which of the following is the likely cause of the loss of instrument air, and what action is required ?

A. Leak in remote location of instrument air system.
Ensure that the containment air system is supplied by the containment makeup system or the containment air compressor.

B. Leak in remote location of instrument air system.
Locate and isolate the leak or scram the reactor and enter OT-3100, Scram Procedure.

C. Clogged air dryer filter.
Bypass the in-service air dryer AND ensure SA-PCV-1 is fully closed.

D. Clogged air dryer filter.
Bypass the in-service air dryer OR cross-connect service air to the instrument air

header.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Indications do not support a leak. (noise, dryer skid indications) If there was a leak, then the action is plausible because it is directed by procedure.

B. Incorrect. Indications do not support a leak. Action is plausible because it would be performed if IA pressure dropped below 55 psig.

C. Incorrect. Correct indications but SA-PCV-1 will not be closed at current IA pressure. It should be throttled open at this pressure.

D. Correct.

Technical Reference(s): OPON-3146-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: Later (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 K5.06	
	Importance Rating	3.0	

Knowledge of the operational implications of the following concepts as they apply to STANDBY LIQUID CONTROL SYSTEM : Tank level measurement

Proposed Question: RO Question # 10

Which one of the following is the operational implication of a loss of instrument air to the Standby Liquid Control (SLC) system?

A. SLC tank level indication will fail upscale.
There will be no other impact on the system.

B. SLC tank level indication will fail downscale.
There will be no other impact on the system.

C. SLC tank level indication will fail downscale.
The tank heater will de-energize if operating.

D. SLC tank level indication will fail upscale.
The pump discharge accumulators will slowly discharge.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Without the instrument air system the level indicator will fail downscale, no d/p to measure.

B. Incorrect - The loss of instrument air fails the SLC tank level instrument downscale since this provides the indication for the low level trip of the SLC tank heater the heater also fails

C. Correct - Instrument air supplies the air for the bubbler dip tube, which is associated with the storage tank level transmitters. Loss of instrument air will cause the local and control room level indicators to fail low. Also the level switch for the heaters will be actuated at 17% indicated level causing a loss of tank heaters in automatic or in manual control

D. Incorrect - Without the instrument air system the level indicator will fail downscale, no d/p to measure. Additionally, the accumulators are charged with nitrogen, not air.

Technical Reference(s): LOT-00-211 pg. 13/19 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-211 K7.09 (As available)

Question Source: Bank # WTSI 12964

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002	K6.05
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM : Reactor water level input

Proposed Question: RO Question # 11

The reactor is operating at 100% power.

The 'A' level instrument is selected to input to the Feedwater Level Control System.

Which of the following describes the 'A' Feedwater Level Control (FWLC) RPV level indicator and FWLC system response to a small break on the 'A' reference leg?

A. Indicated RPV water level will lower.
The Feedwater Regulating Valves will remain as-is.

B. Indicated RPV water level will lower.
The Feedwater Regulating Valves will throttle closed.

C. Indicated RPV water level will rise.
The Feedwater Regulating Valves will remain as-is.

D. Indicated RPV water level will rise.
The Feedwater Regulating Valves will throttle closed.

Proposed Answer: D

Explanation (Optional):

A. Incorrect -the reference leg leak makes indicated level rise, so the FRVs close. FRVs will not remain in position with a failed control signal. Actual level responds by decreasing.

B. Incorrect -the reference leg leak makes indicated level rise, so the FRVs close. Actual level responds by decreasing.

C. Incorrect -the reference leg leak makes indicated level rise, so the FRVs close. FRVs will not remain in position with a failed control signal. Actual level responds by decreasing.

D. Correct Response- the reference leg leak makes indicated level rise, so the FRVs close. Actual level responds by decreasing.

Technical Reference(s): OT-3113 (Attach if not previously provided)

LOT-01-259

DOE FUNDAMENTALS HANDBOOK INSTRUMENTATION AND CONTROL Volume 1 of 2

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-01-259, K3.07 (As available)

Question Source: Bank # VY LOR 1212

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Editorially modified for balance and plausibility

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K6.02	
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the following will have on the RELIEF/SAFETY VALVES : Air (Nitrogen) supply: Plant-Specific

Proposed Question: RO Question # 12

Given the following:

- The plant was operating at full power when a loss of Bus 1 occurred resulting in a manual reactor scram and a valid Group I primary containment isolation signal.
- SRVs have each been cycled by the OATC from CRP 9-3, 5-7 times.
- HPCI was being placed in pressure control by the BOP.

Which one of the following describes the ability of the SRVs to continue to operate under these conditions?

A. SRVs will NOT open in the pneumatic operation mode.

B. SRVs will open ONLY in the self-actuation mode.

C. The SRVs will operate in the pneumatic operation and self-actuation modes.

D. Only SRV-71A and SRV-71C can be manually operated from the RCIC corner room.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Backup nitrogen bottles will supply the motive force to operate all SRVs in the pneumatic operation mode after the N2 accummulators are exhausted. Self-actuation mode will always continue to work.

B. Incorrect – With a loss of Bus 1 then a loss of Bus 11 occurs which removes the normal N2 supply from the drywell and loads in it. The SRV accumulators will allow between 2 and 5 cycles depending on DW pressure conditions. Backup nitrogen bottles will supply the motive force to operate all SRVs in the pneumatic operation mode.

C. Correct - With a loss of Bus 1 then a loss of Bus 11 occurs which removes the normal N2 supply from the drywell and loads in it. The SRV accumulators will allow between 2 and 5 cycles depending on DW pressure conditions. Backup nitrogen bottles will supply the motive force to operate all SRVs in the pneumatic operation mode.

D. Incorrect – This would only be a necessary option for shutdown outside the control room. With a loss of Bus 1 and Bus 11 the pneumatic operation of the valves is not affected provided the backup N2 bottles are in service. There is nothing in the stem to indicate otherwise. 'A' and 'B' SRV's are operated from the alternate shutdown panel not A and C.

Technical Reference(s): LOT-00-218 pg. 23 and 24 (Attach if not previously provided)
ADS Design Basis Document

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-218 EO K6.04 (As available)

Question Source: Bank # WTSI 11126
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Duane Arnold

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

10 CFR Part 55 Content: 55.41 8
55.43
Components, capacity, and functions of emergency systems.
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A1.09	
	Importance Rating	3.0	

Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Maintaining minimum load on emergency generator (to prevent reverse power)

Proposed Question: RO Question # 13

The 'A' EDG has been running loaded to 2700 KW for the Monthly Slow Start Operability test per OPST-EDG-4126-02A.

The surveillance run is completed and steps are being taken to secure the EDG.

When the Operator at CRP 9-8 takes the Diesel GEN SPEED GOVERNOR control switch to LOWER, a switch failure causes 'A' EDG load to rapidly lower to 0 KW.

With no additional Operator action, what is the expected 'A' EDG response to this failure?

A. The EDG Loss of Field relay will actuate on LOW EDG voltage and trip the 86 DG relay resulting in a trip of the 'A' EDG.

B. The 86 DG relay will trip on Generator Overload resulting in a trip of the 'A' EDG.

C. The 86 DG relay will trip on Generator Reverse Power resulting in a trip of the 'A' EDG.

D. The EDG Loss of Field relay will actuate on HIGH EDG voltage and trip the 86 DG relay resulting in a trip of the 'A' EDG.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - GEN SPEED GOVERNOR affects EDG speed and therefor EDG KW load when in isochronous operation. EDG output voltage is unaffected.

B. Incorrect - EDG load will reduce not rise when the GEN SPEED GOVERNOR control switch is taken to LOWER.

C. Correct - Generator load will reduce when the GEN SPEED GOVERNOR control switch is taken to LOWER. This will lead to a Reverse Power trip of the EDG

D. Incorrect - GEN SPEED GOVERNOR affects EDG speed and therefor EDG KW load when in isochronous operation. EDG output voltage is unaffected.

Technical Reference(s): Emergency Diesel Generator And Auxiliary System Design Basis Document (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-264 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New XX

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000	A1.06
	Importance Rating	3.7	

Ability to predict and/or monitor changes in parameters associated with operating the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) controls including: Reactor temperatures (moderator, vessel, flange)

Proposed Question: RO Question # 14

During a refueling outage the following conditions exist:

- The reactor has been shutdown for 20 days.
- The moisture separator is installed.
- Reactor level 190 inches.
- Reactor coolant temperature is 180°F.
- Shut down cooling flow is 4500 GPM.

To prevent thermal stratification under these conditions:

A. Flow must be raised and maintained ≥ 6700 GPM.

B. Reactor coolant temperature must be reduced to ≤ 140°F.

C. Flow must be maintained ≥ 4100 GPM.

D. RPV level must be maintained ≥ 190 inches.

Proposed Answer: C

Explanation (Optional):

A. Incorrect -With RPV level >185 inches we do not have to raise flow to >6700 gpm.

B. Incorrect -Only if RPV level is <185 inches.

C. Correct Response. It has been >15 days since shutdown,

D. Incorrect –The level to maintain is ≥ 185 inches.

Technical Reference(s): OPOP-RHR-2124 (Attach if not previously provided)
GE SIL 357

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-205 K1.02 (As available)

Question Source: Bank # VY LOR 389
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A2.11	
	Importance Rating	3.4	

Ability to (a) predict the impacts of the following on the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Motor operated valve failures
Proposed Question: RO Question # 15

Given the following:

- With the plant at power, a LOCA occurred.
- RHR/LPCI automatically started in the injection mode.
- Reactor pressure is 190 psig and stable.
- RPV level is 15 inches and stable
- Outboard Injection Valve RHR-27B has failed to open, and all manual attempts to open it have failed.

Which ONE of the following identifies the impact of the valve failure; AND the actions that should be taken because of this?

A. ‘B’ and/or ‘D’ RHR pump damage could occur;
Within 1 hour either stop the pumps or raise flow to greater than 4100 gpm.

B. ‘B’ and/or ‘D’ RHR pump damage could occur;
After 1 hour either stop the pumps or raise flow to greater than 4100 gpm.

C. ‘A’ and/or ‘C’ RHR pump damage could occur;
Within 1 hour either stop the pumps or raise flow to greater than 4100 gpm.

D. ‘A’ and/or ‘C’ RHR pump damage could occur;
After 1 hour either stop the pumps or raise flow to greater than 4100 gpm.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect:** This is incorrect because the minimum flow requirement should be 4100

gpm, rather than 2700 gpm. This is plausible because the actual minimum flow requirement stated by the pump vendor is 2700 gpm, with an additional 1400 gpm for instrument uncertainties. The operator may be unaware of the uncertainties penalty.

B. **Correct:** According to OPOP-2124 (p5; Rev 8), the RHR pump vendor has provided pump minimum flow values to establish long term RHR pump minimum flow requirements. The RHR pump vendor indicates that operation of an RHR pump at an indicated flow less than 4100 gpm (flow through the current minimum flow valve is approximately 350 gpm) is allowed for up to four hours. Subsequent to the 4 hour period, flow for each operating RHR pump must be rised to greater than or equal to 4100 gpm. The 4100 gpm value includes the vendor recommended long term minimum flow rate of 2700 gpm to which is added 1400 gpm for instrument uncertainties. The high uncertainty is due to the low operating point relative to the instrument range. Any combination of flow paths may be used to achieve greater than or equal to 4100 gpm pump flow. While the new minimum flow restrictions for the RHR system are always active, under normal conditions, such as full flow testing or shutdown cooling, RHR pump flow is maintained at a value such that long term pump minimum flow requirements are met. According to Attachment 5 of OPOP-2124 (p141; Rev 8), if an RHR pump has operated at less than 4100 gpm for greater than 1 hour, prior to operating at less than 4100 gpm for 4 hours, then verify that the RHR pumps are not required for adequate core cooling and that the RHR pump flow has been less than 4100 gpm for greater than 1 hour. Then, rise RHR pump flow to greater than or equal to 4100 gpm using any combination of flow paths or secure the RHR pump operating at less than 4100 gpm.

C. **Incorrect:** This is incorrect because the B Train is not needed to provide Adequate Core Cooling. According to LOT-00-205 (p55; Rev 33), redundancy and independence are designed into the RHR system to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. This means that adequate core cooling can be achieved by the operation of the A Train alone, and that the B Train is not needed under the given plant conditions. This is plausible because the operator may incorrectly believe that additional injection flow is needed, and if so may conclude that additional flow can be achieved by cross connecting the A and the B trains to use the B and D pumps through the A Train injection lines.

D. **Incorrect:** This is incorrect because the B Train is not needed to provide Adequate Core Cooling. According to LOT-00-205 (p55; Rev 33), redundancy and independence are designed into the RHR system to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. This means that adequate core cooling can be achieved by the operation of the A Train alone, and that the B Train is not needed under the given plant conditions. This is plausible because the operator may incorrectly believe that additional injection flow is needed, and if so may conclude that additional flow can be achieved by aligning the RHRSW Pumps to the A Injection Train.

Technical Reference(s): OPOP-RHR-2124 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-00-205 Objectives K4.02e, K4.03a, K4.05c and K4.05d (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001	A2.09
	Importance Rating	3.1	

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Exceeding voltage limitations

Proposed Question: RO Question # 16

Given the following:

- Both Auto Transformers are deenergized.
- Voltage on Emergency Buses 3 and 4 indicates approximately 3650 VOLTS.
- ISO New England is unsure of how long the condition will last.

Which one of the following actions is required in relation to Bus 3 and 4?

Note: Assume all actions are completed for one bus before completing the same actions for the second bus.

A. IAW ARS 8-J-9, SAFETY BUS VOLTAGE LO, confirm both EDG's start.
Open 3T1/4T2 and allow EDG's to pick up the buses.

B. IAW ARS 8-J-9, SAFETY BUS VOLTAGE LO, confirm both EDG's start.
OPEN Breaker 12 (22) to de-energize Bus 1 (2) initiating a full LNP sequence for each bus.

C. IAW ON-3155, Loss of the Auto Transformer, minimize station electrical loading to the extent possible.
START and PARALLEL the associated diesel generator to its 4 KV bus.
Then separate the bus from the electrical system by opening Breaker 3T1 (4T2).

D. IAW ON-3155, Loss of the Auto Transformer, minimize station electrical loading to the extent possible.
START the associated diesel generator.
Then transfer Bus 3 (4) to the diesel generator by de-energizing the bus by opening Breaker 3T1 (4T2).

Proposed Answer: D

Explanation (Optional):

A. Incorrect –With safety bus voltage >2900 volts, the EDGs will NOT start.The procedure warns not to attempt manually connecting DG B to Bus 3 because of the low voltage condition.

B. Incorrect - With safety bus voltage >2900 volts, the EDGs will NOT start. The sequence/steps are not in accordance with procedural direction and would cause a temporary loss of power and challenge the electrical system.

C. Incorrect -The procedure warns not to attempt manually connecting DG B to Bus 3 because of the low voltage condition.

D. Correct.

Technical Reference(s): ON-3155 pg. 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-601, CRO 3 (As available)

Question Source: Bank # VY LOI 205

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments: VY # 205 editorial mods to make it more plausible

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	2	
Group #	1	
K/A #	218000	A3.05
Importance Rating	3.6	

Ability to monitor automatic operations of the AUTOMATIC DEPRESSURIZATION SYSTEM including: Suppression pool level

Proposed Question: RO Question # 17

- A plant transient has resulted in a torus rupture and ADS actuation.
- Torus level is 5.5 feet and lowering fast.
- RPV pressure is 200 psig and lowering fast.

Which ONE of the following would be the primary containment limit of concern for the above conditions?

A. DWSIL, Drywell spray initiation limit.

B. RPV Saturation temperature limit.

C. PSP, Pressure suppression pressure limit.

D. PCPL-A, Primary containment pressure limit, A.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – With primary containment pressurizing it is likely that the DWSIL will be violated but the primary concern is PCPL-A. EOP-3 will direct primary containment venting to restore pressure and this will likely take pressure to within the safe region of the DWSIL curve.

B. Incorrect – The RPV saturation limit may be violated and cause negative temperature affects to RPV level instrumentation but PCPL-A is the directed limit to comply with IAW EOP-3.

C. Incorrect – PSP is utilized to assure the pressure suppression function of the containment is maintained while either the RPV is at pressure or primary containment flooding is required. With an RPV-ED already in progress and torus level less than 6.8

feet we have already violated PSP.

D. Correct – PCPL-A is a function of primary containment water level. Exceeding the limit may challenge primary containment vent valve operability, SRV operability, or the structural integrity of the primary containment.

Technical Reference(s): EOP-3; EOP Study Guide (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-607 CRO 16, 17 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New XX

Question History:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 A3.01	
	Importance Rating	3.2	

Ability to monitor automatic operations of the D.C. ELECTRICAL DISTRIBUTION including: Meters, dials, recorders, alarms, and indicating lights

Proposed Question: RO Question # 18

Which of the following describes the indications that a loss of 125 DC Bus DC-1 has occurred?

A. Loss of position indication for RCIC System valves and components.
Control Room alarm 4-R-4 ECCS 24 VDC System 'B' Trouble.

B. Loss of position indication for HPCI System valves and components.
Control Room alarms 4-Q-4 ECCS 24 VDC System 'A' Trouble.

C. Loss of position indication for RCIC System valves and components.
Control Room alarms 4-Q-4 ECCS 24 VDC System 'A' Trouble.

D. Loss of position indication for HPCI System valves and components.
Control Room alarm 4-R-4 ECCS 24 VDC System 'B' Trouble.

Proposed Answer: B

Explanation (Optional):

A. Incorrect and plausible alternatives to this question. DC-2/DC-1.

B. Correct as HPCI components are powered from 125 DC bus1 and the power available light will be OFF if there is a loss of this power from DC-1 OR DC-2. DC-1/could be an indication of a loss of DC-2 as well.

C. Incorrect and plausible alternatives to this question. DC-1/DC-2C.

D. Incorrect and plausible alternatives to this question. DC-2A/DC-1C.

Technical Reference(s): OPON-3159-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-263 EO K3.02 and K3.03 (As available)

Question Source: Bank # WTSI 3488

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Monticello

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 A4.07	
	Importance Rating	3.4	

Ability to manually operate and/or monitor in the control room: Verification of proper functioning/ operability

Proposed Question: RO Question # 19

A reactor startup is in progress.
The RO is preparing to withdraw the SRM A and D detectors.

The following conditions exist:

- IRMs indicate midscale on range 4
- SRM ‘A’ indicates 22000 cps; retract permit light illuminated
- SRM ’D’ indicates 18000 cps; retract permit light illuminated

The drive out pushbutton is depressed and held.
‘A’ and ’D’ SRM detectors will begin to withdraw.

Which ONE of the following is the system response?

A. EACH will stop withdrawing when their ASSOCIATED count rate drops below 100 cps.

B: EACH will stop withdrawing when their ASSOCIATED retract permit light extinguishes.

C: BOTH will stop withdrawing when EITHER retract permit light extinguishes.

D: EACH will stop withdrawing when they EACH reach their full out limit.

Proposed Answer: D

Explanation (Optional):

A. Incorrect, the detectors will continue to withdraw until they reach their out limits. A rod block will be inserted at 100 cps. Candidates may select this if they do not recall that the detector drive circuit only responds to a drive command and/or limit switches.

B. Incorrect, the retract permit will only cause a rod block. Candidates may select this if they believe the common misconception that the retract permit signal impacts the ability

to move the detector.

C. Incorrect, the retract permit will only cause a rod block. Candidates may select this if they believe the common misconception that the retract permit signal impacts the ability to move the detector.

D. Correct, the detector drive circuit will only respond to pushbutton commands and/or limit switch actuation.

Technical Reference(s): OP-0105
OP-2130 (Attach if not previously provided)
LOT-01-215

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-01-215 EO K8.02 (As available)

Question Source: Bank # WTSI 14301
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2011 Susquehanna

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004	A4.07
	Importance Rating	3.4	

Ability to manually operate and/or monitor in the control room: Verification of proper functioning/ operability

Proposed Question: RO Question # 20

Which of the following describes the MINIMUM SRM count rate for the SRM to be considered operable; and the basis of this requirement?

A. 3 CPS;
It assures that any transient will begin at or **above** 10^{-8}% CTP, which is the assumed power level used in the analysis of transients from cold conditions.

B. 3 CPS;
It assures that any transient will begin at or **below** 10^{-8}% CTP, which is the assumed power level used in the analysis of transients from cold conditions.

C. 5 CPS;
It assures that any transient will begin at or **above** 10^{-8}% CTP, which is the assumed power level used in the analysis of transients from cold conditions.

D. 5 CPS;
It assures that any transient will begin at or **below** 10^{-8}% CTP, which is the assumed power level used in the analysis of transients from cold conditions.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. Correct value but basis is incorrect. Plausible because the candidate must determine between above and below 10-8 CTP

C. Incorrect. Incorrect value for operability but correct basis. Plausible because 5 CPS is close to 3 CPS and also that CPS is an alarm setpoint

D. Incorrect. Incorrect value and incorrect basis. Plausible because of same reasons as C

and D

Technical Reference(s): TS basis 4.3.B.5 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-01-215 SRO 3 (As available)

Question Source: Bank #
Modified Bank # VY LOI 1856 (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A3.05	
	Importance Rating	3.9	

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: Reactor water level

Proposed Question: RO Question # 21

The plant was operating at 100% power when a transient occurred.

5 minutes after the transient, the following conditions exist:

- RPV water level is 60 inches, lowering at 1 inch per minute.
- Reactor pressure is 600 psig, lowering at 10 psig per minute.
- DW pressure is 4.6 psig; rising at 1 psig per minute.

With these conditions, which ONE of the following describes the status of the following 'A' Core Spray valves to this event?

A. CS Minimum Flow Valve, CS-5A – OPEN.
CS Inboard Isolation Valve, CS-12A – OPEN.
CS Outboard Isolation Valve, CS-11A – OPEN.

B. CS Minimum Flow Valve, CS-5A – CLOSED.
CS Inboard Isolation Valve, CS-12A – OPEN.
CS Outboard Isolation Valve, CS-11A – OPEN.

C. CS Minimum Flow Valve, CS-5A – OPEN.
CS Inboard Isolation Valve, CS-12A – CLOSED.
CS Outboard Isolation Valve, CS-11A – OPEN.

D. CS Minimum Flow Valve, CS-5A – CLOSED.
CS Inboard Isolation Valve, CS-12A – CLOSED.
CS Outboard Isolation Valve, CS-11A – OPEN.

Proposed Answer: C

Explanation (Optional):

A. Incorrect. plausible; would be true if RPV Water Level were was below 82 inches and pressure was <350 psig

B. Incorrect. plausible; would be true if RPV Water Level were was below 82 inches and pressure was <350 psig

C. Correct. with DW pressure >2.5 psig, CS auto initiation occurs, injection valves open with pressure< 350 psig

D. Incorrect. plausible; min flow must be open to provide flow at <600 gpm

Technical Reference(s): LOT-00-209 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-209 K4.08 (As available)

Question Source: Bank # WTSI 11274

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Duane Arnold

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002	2.4.11
	Importance Rating	4.0	

Emergency Procedures / Plan: Knowledge of abnormal condition procedures.

Proposed Question: RO Question # 22

Given the following:

The reactor was at 80% power and holding for a rod pattern adjustment.
Reactor pressure was approximately 970 psig.
Vessel level was 160 inches and steady.
Due to an EPR malfunction, a pressure transient occurred.
The crew then placed the EPR in cutout, but was unable to regain pressure control with the MPR.
The reactor was scrammed.
Plant pressure is now at 790 psig and continuing to lower.
'A' and 'B' main steam line flows indicate 0.7 Mlbm/hour per steam line.
The BOP reports that the #1 bypass valve is indicating open.

The crew should now perform which ONE of the following actions?

A. Take the MPR taken to raise until the bypass valve closes.

B. Send an operator to visually inspect the MHC system to determine why the bypass valve will not close.

C. Shut the MSIVs for the required Group I isolation signal.

D. Trip MTS-2 to close the bypass valves.

Proposed Answer: C

Explanation (Optional):

A. Incorrect -plausible as this is a follow-up action to address lowering pressure if the MPR has failed.
B. Incorrect-Plausible, An operator may eventually be deployed, but it is not an immediate

action, closing MSIVs is required by procedure

C. Correct Response- Operator action required for failure of the Group I isolation signal.

D. Incorrect -Plausible as it may shut the bypass valve, but outside of procedural direction

Technical Reference(s): OT-3115, pg. 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-602 RO EO 5 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000 A1.02	
	Importance Rating	3.8	

Ability to predict and/or monitor changes in parameters associated with operating the STANDBY LIQUID CONTROL SYSTEM controls including: Explosive valve indication
Proposed Question: RO Question # 23

Given the following:

- Reactor power is stable at 19 %.
- A need has arisen to inject the Standby Liquid Control System into the reactor.
- Reactor pressure is 800 psig.
- SLC tank level is 92 %.
- The operator has unlocked the SLC Switch and positioned it to the SYS 1 position.

Two minutes later, the following indications are observed:

- SLC System 1 amber continuity light is NOT LIT.
- CRP 9-5 back panel ammeters read > 0.1 milli amps.
- SLC pump 'A' red status light is LIT, Green status light is NOT LIT.
- SLC pump discharge pressure is approximately 1450 psig.
- The red injection flow indicator light is NOT LIT.
- SLC tank level is at 91% and slowly lowering.
- Reactor power is stable at 19%.

Assuming no instruments failures have occurred, which ONE of the following describes the operational status of the system?

SLC SYS 1 is:

A. operating and injecting as expected.

B. <u>not</u> injecting because the squib valve has not fired, and a leak has developed in the system.

C. <u>not</u> injecting even though the squib valve has fired, and a leak has developed in the system.

D. operating and injecting but at a lower flow rate than expected.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect:** This is incorrect because three of the six indications (SLC Pump discharge pressure, The Red injection flow indicator light, Reactor power is stable) are not as expected. This is plausible because the operator may incorrectly believe that these parameters are normal for system operation (i.e. Pump discharge pressure is expected to be higher than reactor pressure and it is. The amber light goes DARK to function, the operator may believe the Red light does as well. The operator may incorrectly believe that there is significant delay time between the time of injection and the time reactor power is actually lowerd).

B. **Correct:** The SLC system is not injecting due to several stem indications. With SLC pump discharge pressure at ~1450 psig it indicates that the squib has not fired and the system is running, at least in part, through the relief valve. The indication of the continuity light not being lit is a distractor and tests system knowledge. The stem information of continuity meter indication also tests system knowledge and gives the vital piece of information that the system is not injecting. The flow light not lit is important and is sometimes confused with the continuity light only working when the system control switch is returned to neutral. The system has a leak due to slowly lowering tank level.

C. **Incorrect:** This is incorrect because there is no indication that the squib valve has fired. With the switch still in system 1 as per the stem the operator recieves no indication of squib valve status until the switch is returned to the neutral position and the lights checked. Lacking this, the operator must rely on the continuity meter on the 9-5 back panel.

D. **Incorrect:** This is incorrect because power is NOT decreasing. According to Appendix B of OP-2114 (p1 of 1; Rev 36), the operator is directed to ensure that reactor power lowers as the SLC fluid is injected (A sure sign that injection is occurring). This is plausible because according to LOT-00-211 (p17; Rev 17), the tank volume results in 48.3 gallons per %indicated level. If the level is dropping at 1% in in two minutes, this is a volume loss of 48.3 gallons, or about 24 gpm, which is consistent with the Red flow indicating light being DARK.

Technical Reference(s): LOT-00-211 (p17 and 20; Rev 17) (Attach if not previously provided)
Appendix B of OP-2114 (p1 of 1;
Rev 36)
P&ID G-191171

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-00-211 EO K4.08 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K4.02	
	Importance Rating	3.1	

Knowledge of D.C. ELECTRICAL DISTRIBUTION design feature(s) and/or interlocks which provide for the following: Breaker interlocks, permissives, bypasses and cross ties: Plant-Specific

Proposed Question: RO Question # 24

Given the following:

- The plant is at rated power.
- A loss of DC-3 has occurred.

Which of the following describes the effect, if any, on DC-3A?

A. No effect as DC-3A will remain energized.

B. DC-3A will remain energized.
Alternate supply to DC-3A is lost.

C. DC-3A is deenergized.
Power must be restored to DC-3 to reenergize DC-3A.

D. DC-3A is deenergized.
A mechanical interlock on DC-3A will allow transfer of DC-3A to DC-1.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. A loss of DC-3 will cause loss of normal supply to DC-3A. Alternate supply comes from DC-1.

B. Incorrect. DC-3 does not have to be energized to restore DC-3A because DC-1 can provide power through interlock. Alternate supply comes from DC-1 which was not lost.

C. Incorrect. DC-3 does not have to be energized to restore DC-3A because DC-1 can provide power through interlock.

D. Correct.

Technical Reference(s): OP-2145
ON-3160 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-263 EOK4.02 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001	K1.02
	Importance Rating	3.3	

Knowledge of the physical connections and/or cause- effect relationships between A.C. ELECTRICAL DISTRIBUTION and the following: D.C. electrical distribution

Proposed Question: RO Question # 25

With the plant at 100% power, a loss of 125V Bus DC-2 occurs.

Can Bus 6 and Bus 7 be cross-connected under these conditions, and why or why not?

A. Yes, breaker 6T7 can be closed because its control power is supplied by DC-1.

B. Yes, breaker 6T7 can be closed because its control power is supplied by DC-3.

C. No, breaker 6T7 can NOT be closed until its control power is restored from DC-3A.

D. No, breaker 6T7 can NOT be closed until its control power is restored from DC-3B.

Proposed Answer: D

Explanation (Optional):

A. Incorrect -6T7 breaker receives control power from DC-3B.

B. Incorrect -6T7 breaker receives control power from DC-3B.

C. Incorrect -DC-3A powers up control room annunciators.

D. Correct.

Technical Reference(s): OP 2143; OP 2145; ON 3160 (Attach if not previously provided)
P&ID's G-191372-SH-01 & 2

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-02-262 Obj 3 (As available)

Question Source: Bank # VY LOR 1364

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 A4.01	
	Importance Rating	2.8	

Ability to manually operate and/or monitor in the control room: Transfer from alternative source to preferred source

Proposed Question: RO Question # 26

Given the following:

- The plant was operating at 100% power when a Loss of Normal Power (LNP) occurred.
- The 'B' EDG failed to start but the 'A' EDG operatesd as designed.
- RPV water level is being controlled in the normal range by HPCI following the scram after being started manually.
- No accident signals are present.

Five minutes after the event, which one of the following would be correct with respect to 'A' UPS and MCC-89A?

____1.____ driving the AC generator and the Maintenance Tie ___2.____.

A. 1. DC motor
   2. available

B. 1. DC motor
   2. unavailable

C. 1. AC motor
   2. available

D. 1. AC motor
   2. unavailable

Proposed Answer: C

Explanation (Optional):

A. Incorrect - RUPS automatically shifts back to the AC motor

B. Incorrect - RUPS automatically shifts back to the AC motor. Bus 9 also supplies MCC 9B which supplies the Maintenance Tie

C. Correct - A full LNP causes bus 9 to deenergize. Bus 9 feeds the 'A' RUPS AC motor. The motor shifts to the DC motor when it reaches < 85% voltage. The 'A' EDG starts and supplies Bus 9 within 13 secs. The AC motor starts again (its breaker never trips) and RUPS automatically shifts back to the AC motor. Bus 9 also supplies MCC 9B which supplies the Maintenance Tie

D. Incorrect - Bus 9 supplies MCC 9B which supplies the Maintenance Tie.

Technical Reference(s): OP 2144 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-03-262, CRO 7 (As available)

Question Source: Bank # WTSI 3791

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 Vermont Yankee

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	226001 K1.12	
	Importance Rating	3.0	

Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following: Suppression pool (spray penetration): Plant-Specific.

Proposed Question: RO Question # 27

Given the following:

- A small recirc leak in containment caused drywell pressure to rise.
- A manual reactor scram was inserted.
- EOP-3 has been entered.
- Torus spray is in service using the 'A' loop of RHR.
- The CRS has directed drywell sprays to be placed in service on the 'A' loop of RHR to address 10 psig torus pressure and rising.

Which ONE of the following describes the number of 'A' loop RHR pumps used and the reason for the alignment while initiating DW spray?

A. Two RHR pumps are used to maximize the rate of pressure reduction.

B. Two RHR pumps are used to ensure torus temperature remains within limits.

C. One RHR pump is used due to the heat exchanger tube size restriction.

D. One RHR pump is used to ensure that a second RHR pump is available for future use.

Proposed Answer: C

Explanation (Optional):

A: Incorrect. Only 1 RHR pump in the 'A' loop is used with other RHR pump in PTL. Plausible because Torus spray is designed to reduce pressure

B: Incorrect. Only 1 RHR pump in the 'A' loop is used as above, and plausibility is introduced because Torus temperature will be reduced by RHR spray.

C: Correct. Because RHR pumps provide high flow, using more than 1 pump per loop for the size of the heat exchanger tubes would not result in significantly raised flow but could cause vibration damage to the heat exchznger tubes.

D: Incorrect. While one RHR pump per loop is placed in PTL, implying it is to remain available, this is not the reason for the action

Technical Reference(s): OPOP-RHR-2124 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-205 K1.03e and K1.04e (As available)

Question Source: Bank # WTSI 14703

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Columbia

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001	K2.01
	Importance Rating	3.3	

Knowledge of electrical power supplies to the following: Reactor Feedwater pumps: motor drive only

Proposed Question: RO Question # 28

The plant was operating at 80% RTP with the Feedwater System in the following lineup:

- RFPs ‘A’ and ‘B’ running.
- RFP ‘C’ in Standby.
- The P-1-1B (RFP ‘B’) TRIP ON COND PUMP TRIP control switch is in BYPASS.

Subsequently, the plant experiences a loss of 4KV Bus 2.

Which ONE of the following statements describes the response of the Feedwater Pumps 10 seconds after the loss of Bus 2?

(NOTE: Assume **no** operator action is taken.)

A. Feed Pump A trips.
Feed Pump B continues to run.
Feed Pump ‘C’ remains off.

B. Feed Pump A continues to run.
Feed Pump B continues to run.
Feed Pump ‘C’ remains off.

C. Feed Pump A continues to run.
Feed Pump B trips.
Feed Pump ‘C’ remains off.

D. Feed Pump A continues to run.
Feed Pump B trips.
Feed Pump C breaker closes and remains closed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Feed Pump A will not receive a trip signal, only Feed Pump B will

B. Incorrect - Feed Pump B will receive a trip signal on the loss of Bus 2

C. Correct.

D. Incorrect - Feed Pump C breaker will not remain closed with no power on Bus 2

Technical Reference(s): OP 2172, OT 3170 (Attach if not previously provided)
P&ID G-191299

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-259 K1.07, K2.01 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201002 K3.02	
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: Rod block monitor: Plant-Specific
Proposed Question: RO Question # 29

Given the following:

- The plant is operating at 35% power.
- A failure occurs in the Reactor Manual Control System such that the signal to the Select Matrix indicates that more than one Rod is selected.

Which ONE of the following describes how the Rod Block Monitors will be affected by this failure?

A. An Inop Trip will occur.

B. A High Trip will occur.

C. A Downscale Trip will occur.

D. No effect, the RBMs are bypassed at this power level.

Proposed Answer: A

Explanation (Optional):

A. **Correct:** According to Transparency 1 of LOT-03-201 (Rev 18) when no rod is selected or more than one rod is selected in the Select Matrix, an Inop Trip occurs.
B. **Incorrect:** This is incorrect because according to LOT-03-201 (p17; Rev 18), a high level trip will only occur if the local averaging thermal power from the averaging circuit equals the trip level setpoint generated by the trip reference level select circuit. This is plausible because it is one of three different RBM trips, and the operator may not know what causes a high level trip.
C. **Incorrect:** This is incorrect because according to LOT-03-201 (p17; Rev 18),a downscale trip will only occur if the local average thermal power signal drops to 5%. This is plausible because it is one of three different RBM trips, and the operator may not

know what causes a Downscale trip.

D. **Incorrect:** This is incorrect because the power level is too high to bypass both RBMs. This is plausible because according to LOT-03-201 (p25; Rev 18), an RBM channel will automatically go out of service when its reference APRM indicates less than 30% power. The operator may not know the bypass threshold.

Technical Reference(s): Transparency 1 of LOT-03-201 (Rev 18) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-02-201 Objective K3.02 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000	K4.03
	Importance Rating	2.9	

Knowledge of REACTOR WATER CLEANUP SYSTEM design feature(s) and/or interlocks which provide for the following: Over temperature protection for system components

Proposed Question: RO Question # 30

Given the following:

- A plant startup and heatup is in progress.
- Reactor coolant temperature is 400°F.

The following alarm is received in the control room:

- RWCU DEMIN INLET TEMP HI ISOL, 4-J-2

Which one of the following describes (1) the setpoint, and (2) a potential cause of this alarm?

A. (1) 125°F
(2) Insufficient Drain flow rate.

B. (1) 140°F
(2) Insufficient Drain flow rate.

C: (1) 125°F
(2) Excessive Drain flow rate.

D: (1) 140°F
(2) Excessive Drain flow rate.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. 125 F is setpoint for RWCU area temperature alarm that would provide indication of a line rupture in RWCU area. Insufficient drain flow would result in lower

demin inlet temperature as RBCCW would provide relatively more cooling to drain flow

B. Incorrect. Correct temperature, incorrect cause, as in A

C. Incorrect. Correct cause, but 125 F is the setpoint for the RWCU area alarm

D. Correct. Sertpoint is 140 F for NRHX Outlet/Demin inlet. System isolates inlet and return valves and trips operating RWCU pump. Alarm response indicates if drain flow regulator setting is too high, this alarm may be received

Technical Reference(s): 4-J-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-204 Obj A2.12, A2.13 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	268000 K5.02	
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to RADWASTE : Radiation hazards and ALARA concept
Proposed Question: RO Question # 31

Given the following:

A loss of reactor coolant has resulted in a need to vent the primary containment using Appendix HH of OE 3107.

The vent path is being aligned by operations personnel through radwaste to the primary vent stack when, Radwaste Building Hi Radiation (3-F-4), comes into alarm.

Which one of the following describes actions that are appropriate for the given conditions?

A. Evacuate all personnel from Radwaste and stop the primary containment venting.

B. Evacuate all non-essential personnel from Radwaste and stop the primary containment venting.

C. Evacuate all personnel from Radwaste and continue the primary containment venting.

D. Evacuate all non-essential personnel from Radwaste and continue the primary containment venting.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect:** Operations personnel are considered essential personnel during emergency situations and should not be evacuated. The primary containment venting is required by OE 3107, EOP/SAG appendices and therefore should continue.
B. **Incorrect:** Operations personnel are considered essential personnel during emergency situations. The primary containment venting is required by OE 3107, EOP/SAG appendices and therefore should continue.
C. **Incorrect:** Operations personnel are considered essential personnel during emergency situations and if evacuated, they will not ba able to align the system for venting.

D. **Correct:** Operations personnel are considered essential personnel during emergency situations. The primary containment venting is required by OE 3107, EOP/SAG appendices and therefore should continue.

Technical Reference(s): OPON-3153-01 (Attach if not previously provided)

3-F-4 ARS

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-00-266 Objective K5.02 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	288000	K6.02
	Importance Rating	2.5	

Knowledge of the effect that a loss or malfunction of the following will have on the PLANT VENTILATION SYSTEMS : Applicable component cooling water system: Plant-Specific

Proposed Question: RO Question # 32

Given the following:

- A pipe rupture has resulted in a complete loss of Service Water.
- All automatic actions have occurred.

Which of the following describes how this event ultimately affects the Reactor Recirculation Units (RRUs) and the Reactor Building Air Conditioners (RBAC) 1A - 1D?

The cooling capability of RRUs will be ___ (1) ___.
The cooling capability of RBACs will be ___ (2) ___.

A. (1) lost
(2) lost

B. (1) lost
(2) unaffected

C. (1) unaffected
(2) lost

D. (1) unaffected
(2) unaffected

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect. RBACs will be affected because Service Water chills the refrigerant, and the compressor motor will shut off as refrigerant pressure rises due to the loss of cooling water supply

C. Incorrect. Service Water supplies the cooling coil to RRUs. A candidate can mistakenly believe that CCW supplies cooling since it is in containment.

D. Incorrect. See B and C above

Technical Reference(s): LOT-01-288 (Attach if not previously provided)
ON-3148 K6.02
P&ID G-191159

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-01-288 Obj (As available)

Question Source: Bank #
Modified Bank # WTSI 1258 (Note changes or attach parent)
New

Question History: Last NRC Exam: 2004 River Bend

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	234000 A1.03	
	Importance Rating	3.4	

Ability to predict and/or monitor changes in parameters associated with operating the FUEL HANDLING EQUIPMENT controls including: core reactivity level

Proposed Question: RO Question # 33

During refueling operations with the platform over the core, control rod 22-11 is removed for replacement.

What is the effect on net core reactivity and shutdown margin (SDM)?

A. Net core reactivity lowers;
SDM lowers

B. Net core reactivity rises;
SDM rises

C. Net core reactivity lowers;
SDM rises

D. Net core reactivity rises;
SDM lowers

Proposed Answer: D

Explanation (Optional):

A. plausible; may mis-understand the relationship between net reactivity & SDM

B. plausible; may mis-understand the relationship between net reactivity & SDM

C. plausible; may mis-understand the relationship between net reactivity & SDM

D. Correct – removal of control rod blade increase core reactivity. SDM is decreased

Technical Reference(s): VY Technical Specifications Section 4.1 & 4.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-234, Refueling Operations A1.03 (As available)

Question Source: Bank # WTSI 1300

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Fermi

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	230000 A2.12	
	Importance Rating	3.2	

Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve logic failure
Proposed Question: RO Question # 34

Given the following:

- The plant was operating at power.
- Torus cooling was in service on the 'A' loop of RHR in preparation for a HPCI surveillance.

Subsequently, a LOCA occurs and RHR-34A and 39A (Torus Spray/ Cooling Valves) fail to automatically close as required.

Which ONE of the following identifies the operational concern with these failures, AND the MINIMUM action needed to mitigate THIS concern?

A. LPCI flow is bypassing the core.
Both valves must be manually closed.

B. LPCI flow is bypassing the core.
At least one of the two valves must be manually closed.

C. RHR/LPCI Pump runout will occur.
Both valves must be manually closed.

D. RHR/LPCI Pump runout will occur.
At least one of the two valves must be manually closed.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect:** 1st part correct, 2nd part wrong. Both valves will be closed procedurally (Attachment 5 of OPOP-2124), but only one valve needs to be closed to stop water diversion.

B. **Correct:** 1st part correct, 2nd part correct. The purpose of the interlock to close them is to ensure that water is not diverted from the core during the injection phase. The valves are in series closing one of the two valves will divert all water from Torus Spray into the core through the injection valves.

C. **Incorrect:** 1st part wrong, 2nd part wrong. See A and D.

D. **Incorrect:** 1st part wrong, 2nd part correct. There are no runout concerns listed in procedure. Plausible because condition will create high pump flow which leads to runout.

Technical Reference(s): LOT-00-205 (p35; Rev 33) (Attach if not previously provided)

Residual Heat Removal System
Design Basis Document

Proposed References to be provided to applicants during examination:

Learning Objective: LOT-00-205 Objective K4.15a and K4.03.c (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215002 A3.04	
	Importance Rating	3.6	

Ability to monitor automatic operations of the ROD BLOCK MONITOR SYSTEM including:
Verification or proper functioning/ operability: BWR-3,4,5
Proposed Question: RO Question # 35

Given the following:

- The plant is operating at 100% power.
- Both Rod Block Monitor (RBM) A and B are aligned per plant procedures.
- Rod 26-23 is selected.
- There are no RBM alarms on CRP 9-5.

Which ONE of the following identifies the expected condition of the four status lights on CRP 9-5 and the RBM Meter on CRP 9-14, for each RBM?

A. All RBM status lights on CRP 9-5 are NOT LIT.
The meter on CRP 9-14 indicates the average of the LPRMs assigned to that channel.

B. All RBM status lights on CRP 9-5 are LIT.
The meter on CRP 9-14 indicates the average of the LPRMs assigned to that channel.

C. All RBM status lights on CRP 9-5 are LIT.
The meter on CRP 9-14 indicates the value of the APRM being used as the RBM reference.

D. All RBM status lights on CRP 9-5 are NOT LIT.
The meter on CRP 9-14 indicates the value of the APRM being used as the RBM reference.

Proposed Answer: A

Explanation (Optional):

A. **Correct:** 1st part correct, 2nd part correct. OP-0150 control panel walkdown checklists states RBM Channel lights are OFF, and Channel switches in Average.

B. **Incorrect:** 1st part wrong, 2nd part correct. Lights are only on when condition exists (Hi, Downscale, Inop, Bypass).

C. **Incorrect:** 1st part wrong, 2nd part wrong. See B and D.

D. **Incorrect:** 1st part correct, 2nd part wrong. Meter would indicate this if switch was in APRM REF.

Technical Reference(s): OP-0150 (p15 & 43; Rev 187) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-03-201 K1.14 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290003	A4.01
	Importance Rating	3.2	

Ability to manually operate and/or monitor in the control room: Initiate/reset system

Proposed Question: RO Question # 36

Given the following:

- The Control Room Ventilation System has tripped, following a loss of 480 VAC Bus 8.
- All actions for Loss of Bus 8 have been completed up to the point where Control Room Ventilation is being restarted.
- Power has been restored to 480 VAC Bus 8.

(1) Which one of the following actions is required before Control Room Ventilation can be returned to service?

AND

(2) What are the Technical Specifications (TS) implications, if any, if the control room ventilation system cannot be re-started?

A. (1) PCIS Group 3 isolation must be reset.
(2) There are no associated TS for loss of Control Room Ventilation System.

B. (1) PCIS Group 3 isolation must be reset.
(2) Loss of Control Room Ventilation System causes a loss of Battery Room. Ventilation (SEF-3) which is required by TS.

C. (1) Depress “Drywell Clg and Ctrl Room A/C Blocking Reset” pushbutton on CRP 9-25.
(2) There are no associated TS for loss of Control Room Ventilation System.

D. (1) Depress “Drywell Clg and Ctrl Room A/C Blocking Reset” pushbutton on CRP 9-25.
(2) Loss of the Control Room Ventilation System causes a loss of Battery Room. Ventilation (SEF-3) which is required by TS.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - The Drywell Clg and Ctrl Room A/C Blocking Reset pushbutton must be depressed prior to starting the coolers. A loss of Bus 8 does cause a group 3 isolation. The Control Room Ventilation System supplies Battery Room Ventilation which is required by TS 3.10.2.a

B. Incorrect - The Drywell Clg and Ctrl Room A/C Blocking Reset pushbutton must be depressed prior to starting the coolers, a loss of Bus 8 does cause a group 3 isolation.

C. Incorrect - The Control Room Ventilation System supplies Battery Room Ventilation which is required by TS 3.10.2.a

D. Correct – The Drywell Clg and Ctrl Room A/C Blocking Reset pushbutton must be depressed prior to starting the coolers following the restoration of power from Bus 8. The Control Room Ventilation System supplies Battery Room Ventilation which is required by TS 3.10.2.a

Technical Reference(s): OP 2192, sect. I (Attach if not previously provided)

T.S. 3.10.2.a

Proposed References to be provided to applicants during examination: NO

Learning Objective: (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:2009 VY audit #37

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290001 2.2.44	
	Importance Rating	4.2	

Low Pressure Core Spray: Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions.
Proposed Question: RO Question # 37

Given the following:

- A plant transient has resulted in the Core Spray system initiation signal.
- Both Core Spray pumps have started.
- Reactor pressure is 100 psig.
- Core Spray pump 'A' is operating at 3000 gpm. on CRP 9-3 flow indicator.
- Core Spray pump 'B' is operating at 2700 gpm. on CRP 9-3 flow indicator.

Which ONE of the following identifies the CS loop with the failed open minimum flow valve, AND how the CS-12 valves would respond to the operator taking their control switches to the CLOSE position?

A. Loop 'A' has a failed open Minimum Flow Valve.
The CS-12 Valves will throttle closed.

B. Loop 'A' has a failed open Minimum Flow Valve.
The CS-12 Valves will remain fully open.

C. Loop 'B' has a failed open Minimum Flow Valve.
The CS-12 Valves will throttle closed.

D. Loop 'B' has a failed open Minimum Flow Valve.
The CS-12 Valves will remain fully open.

Proposed Answer: C

Explanation (Optional):

A. **Incorrect:** 1st part wrong, 2nd part correct. This would be true if the system flow indicators were opposite of what they are. Lower flow indicates failure

B. **Incorrect:** 1st part wrong, 2nd part wrong. The CS-12 valves will re-open once throttled and the initiation signal clears and comes in a second time.

C. **Correct:** 1st part correct, 2nd part correct. The system flow indicators are downstream of the minimum flow tap-off, and will indicate a loss of flow, rather than additional flow through the minimum flow line. The CS-12 valves have throttle capability even with the initiation signal still present.

D. **Incorrect:** 1st part correct, 2nd part wrong. See B and C.

Technical Reference(s): Appendix B of OP-2123 (Rev 45) (Attach if not previously provided)
LOT-00-209 Transparency 1 (p23; Rev 28)
P&ID G-191168
Core Spray Design Basis Document

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-00-209 Objective K4.05 and K7.06 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 K5.02	
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to CONTROL ROD DRIVE HYDRAULIC SYSTEM : Flow indication

Proposed Question: RO Question # 38

The reactor scrammed two minutes ago.
The RO notes the CRD system flow control valve is closed but CRD system flow indication is off scale high.

Under these conditions, the RO should perform which ONE of the following?

A. Inform the CRS and have an AO investigate the faulty valve position indication.

B. Trip the operating CRD pump as soon as all control rods are verified to be fully inserted to prevent pump runout.

C. Take manual control of the flow control valve and reduce CRD system flow to approximately 65 gpm.

D. Continue with scram actions since the CRD indications are normal.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Condition is normal. There is no need to investigate.
Plausible: Action from off normal procedure in response to valve failure

B. Incorrect: There is no need to trip the pump. The system's orifice is designed to prevent runout during a scram condition.
Plausible: Action to prevent pump damage

C. Incorrect: Condition is normal. There is no need to take manual control.
Plausible: Action from off normal procedure in response to controller failure

D. Correct: Charging header flow will cause indication to be high, until the SCRAM is reset.

Technical Reference(s): VY Control Rod Drive System Design Basis Document (Attach if not previously provided)
P&ID G-191170

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-01-201 Control Rod Hydraulic System K5.01 (As available)

Question Source: Bank # WTSI 1613
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2007 Hope Creek

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

10 CFR Part 55 Content: 55.41 10
55.43
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021	AK1.01
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as they apply to LOSS OF SHUTDOWN COOLING : Decay heat

Proposed Question: RO Question # 39

Given the following:

- The plant is in cold shutdown, twenty-four hours after reactor shutdown, following extended full power operation.
- 'B' Residual Heat Removal (RHR) pump is operating in the Shutdown Cooling Mode.
- Reactor Coolant Temperature is 135°F on a very slow downward trend.
- No Reactor Recirculation pumps are in service.
- Reactor water level is being maintained at 190 inches.
- MSIVs are shut.

Which of the following describes the long term (greater than one hour) Reactor coolant temperature response if the 'B' RHR pump trips?

Assume no operator action is taken.

Coolant temperature will:

A. Lower until equilibrium is reached in the RHR heat exchanger.

B. Rise until bulk boiling occurs, and reactor pressure rises above atmospheric pressure.

C. Rise until bulk boiling occurs, with reactor pressure steady at atmospheric pressure.

D. Lower until Reactor Coolant Temperature is equal to Service Water Temperature in the RHR heat exchanger.

Proposed Answer: B

Explanation (Optional):

A: Incorrect - With the RHR pump tripped there is no longer shutdown cooling flow from the reactor vessel to the RHR heat exchanger.

B: Correct - Decay heat will cause RPV coolant temperature to rise and eventually reach boiling. Reactor pressure will rise above atmospheric pressure (NOTE: Even if examinee assumes RPV head vents are open pressure will still rise since the head vents are on a 2" line and are designed for removal of non-condensibles at power or air removal for refueling or hydro test conditions. There is industry OE that confirms that bulk boiling of coolant due to lack of shutdown cooling will result in going greater than 212 F and pressurizing the RPV with the vents open).

C: Incorrect - Reactor pressure will rise above atmospheric.

D: Incorrect - With the RHR pump tripped there is no longer shutdown cooling flow from the reactor vessel to the RHR heat exchanger.

Technical Reference(s): ON-125, GP-12
OP 0105 (Attach if not previously provided)
OPOP-RHR-2124
Generic Decay Heat Curve

Proposed References to be provided to applicants during examination: No

Learning Objective: LOT-00-205 K3.01.b & K3.03.b (As available)

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

Question History: 2007 Peach Bottom

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

10 CFR Part 55 Content: 55.41 14
55.43

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003	AK1.05
	Importance Rating	2.6	

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Failsafe component design

Proposed Question: RO Question # 40

The plant was operating at 90% power.
The breaker for the 'B' SBGT valves solenoids has tripped open.

Which ONE of the following describes the correct effect on the system due to the loss _____1____ _?

and the reason for these valve positions____2_____.

A. 1. SGT-1B, 2B, 3B fail open
SGT-4B fails closed

2. Align SBGT for normal plant conditions

B. 1. SGT-1B, 2B, 3B fail closed
SGT-4B fails open

2. Align SBGT for accident conditions

C. 1. SGT-1B, 2B, 3B fail open
SGT-4B fails closed

2. Align SBGT for accident conditions

D. 1. SGT-1B, 2B, 3B fail closed
SGT-4B fails open

2. Align SBGT for normal plant conditions

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The valve position failure to open is correct but the reason is not to support normal operations but provide a vent path for accident conditions.
B. Incorrect – The valve(s) positions are incorrect as 1-3B should be open for providing a vent path. The reason is correct.
C. Correct Response: The valve position failure to open is open for providing a vent path during accident conditions.
D. Incorrect - The valve(s) positions are incorrect as 1-3B should be open for providing a vent path. The reason is incorrect as we are not concerned with normal operations but accident conditions.

Technical Reference(s): OP 3146, Rev. 15 (Appendix B, pg. 2) (Attach if not previously provided)
P&ID G-191175

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-261 k6.06 (As available)

Question Source: Bank # VY LOR 2254
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference: Level RO SRO

Tier #	1	
Group #	1	
K/A #	295026	EK1.02
Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE : Steam condensation

Proposed Question: RO Question # 41

Given the following:

- Torus Water Temperature is rising due to a leaking SRV.

The FIRST operational concern that will be reached is based on the Torus having the inability to provide which ONE of the following?

A. Adequate NPSH to core spray pumps.

B. Complete steam condensation following a LOCA.

C. Adequate cooling to the HPCI turbine oil cooler.

D. Complete steam condensation following an SRV actuation.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. plausible; would be true for higher Torus Water Temperatures

B. Correct. 90 F is the first operational limit reached. This value is based on complete steam condensation following a Loss of Coolant Accident, which will limit maximum post-LOCA Containment Pressure to 62 psig

C. Incorrect. plausible; would be true for higher Torus Water Temperatures

D. Incorrect. plausible; would be true for low Torus Water Level conditions

Technical Reference(s): VY FSAR Section 14.6.3.1.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-223 K4.01 (As available)

Question Source: Bank # WTSI 11300
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2007 Duane Arnold

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

10 CFR Part 55 Content: 55.41 8
55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037	EK2.04
	Importance Rating	4.4	

Knowledge of the interrelations between SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN and the following: SBLC system

Proposed Question: RO Question # 42

Given the following plant conditions:

- The plant has experienced an ATWS from 100% power.
- EOP-2 is being implemented with reactor power at approximately 22%.
- OE 3107 rod insertion appendices are NOT working.
- SLC system 'A' was started at 1230 hours.

With SLC injecting at the Tech Spec pump flow rate, at what time can a plant cooldown be commenced on boron injection alone?

(Note: OP 4114, Table 1 reference provided)

A. 1251

B. 1258

C. 1306

D. 1311

Proposed Answer: D

Explanation (Optional):

A. Incorrect :21 minutes based on a previous VY number of 15% for HSBW

B. Incorrect: This is the time to inject hot shutdown boron weight

C. Incorrect: If candidate incorrectly interprets what Tech Spec injection rate is and uses 40 GPM versus 35.

D. Correct Response: 30% tank level change(CSBW) X 48.3 divided by 35 gpm = 41.4 minutes

Technical Reference(s): VY Technical Specifications 4.4.A,
EOP-2
OP 4114 figure 1 48.3 gals = 1%
Cold S/D boron weight = 30%
EOP 2
(Attach if not previously provided)

Proposed References to be provided to applicants during examination: OP 4114, Table 1

Learning Objective: LOT-00-211 CRO 4 (As available)

Question Source: Bank # VY LOR 2049
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 EK3.02	
	Importance Rating	3.9	

Knowledge of the reasons for the following responses as they apply to HIGH REACTOR PRESSURE : Recirculation pump trip: Plant-Specific

Proposed Question: RO #43

Following a Main Turbine trip, reactor pressure peaked at 1180 psig and reactor water level lowered to -60 inches and is slowly recovering.

While responding to the transient you report that both Reactor Recirculation pumps are running.

The Control room supervisor directs you to trip both recirc pumps because the high reactor pressure and low-low level condition should have caused __(1)__?

Tripping the pumps will add negative reactivity to counteract the positive reactivity addition caused by the __(2)__?

A. (1) ONLY the Recirc Pump generator field breakers to trip

(2) initial pressure spike

B. (1) ONLY the Recirc Pump drive motor breakers to trip

(2) pressure lowering following the scram

C. (1) each Recirc Pump drive motor breaker AND generator field breaker to trip

(2) initial pressure spike

D. (1) each Recirc Pump drive motor breaker AND generator field breaker to trip

(2) pressure lowering following the scram

Proposed Answer: C

Explanation (Optional):

A: Incorrect. Drive motor breakers also trip

B: Incorrect. Generator field breakers also trip

C: Correct. Both breakers trip, pump trip adds - reactivity to counteract positive reactivity from pressure spike

D: Incorrect. Pressure decrease does not add positive reactivity

Technical Reference(s): (Attach if not previously provided)

Proposed References to be provided to applicants during examination: No

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: 2011 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AA1.01	
	Importance Rating	3.6	

Ability to operate and/or monitor the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Grid frequency and voltage.

Proposed Question: RO Question # 44

Procedure OPON-3179-01, Grid Instabilities, has been entered following a report from ISO NE of unstable grid conditions.

Current generator conditions are:

- 640 Mwe.
- MVARs oscillating between 25 MVARs IN and 50 MVARs OUT.
- The main generator voltage regulation system is aligned for normal full power operation.

ISO NE directs that the existing stability limits require that Vermont Yankee adjust grid voltage by changing the amount of reactive loading to 100 MVARs OUT.

This is accomplished by going to …..

A. RAISE on the DC Voltage adjust.

B. RAISE on the AC Voltage adjust.

C. LOWER on the DC Voltage adjust.

D. LOWER on the AC Voltage adjust.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The normal full power configuration for the main generator is with voltage regulation in automatic. This places the AC regulator in service. Going to raise on the DC voltage adjust will have no effect on VARs and system voltage.

B. Correct: The normal full power configuration for the main generator is with voltage regulation in automatic. This places the AC regulator in service. Going to raise on the AC voltage adjust will raise the amount of excitation and cause VARs to go from zero MVAR to the out direction.

C. Incorrect: Lowering the DC voltage adjust will have no effect on VARs and system voltage.

D. Incorrect: Lowering the AC voltage adjust will reduce excitation further tending to reduce generator voltage and causing VARs to move even further in the leading direction.

Technical Reference(s): Simulator (Attach if not previously provided)

LOT-00-304

Proposed References to be provided to applicants during examination: None

Learning Objective: Lesson plan LOT-02-245, RO EO #3. (As available)

Question Source: Bank # WTS Bank # 11876

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 NMP2

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AK3.01	
	Importance Rating	3.8	

Knowledge of the reasons for the following responses as they apply to SCRAM : Reactor water level response

Proposed Question: RO Question # 45

Complete the following statement describing the IMMEDIATE reactor level response to a manual scram from rated conditions, AND the reason for the response?

Reactor level will...

A. lower due to the Recirc pump runback.

B. rise due to delay in Bypass Valves opening.

C. lower due to collapsing voids in the core region.

D. rise due to rapid closure of the Turbine Control Valves.

Proposed Answer: C

Explanation (Optional):

A. Plausible – misconception on increasing / decreasing recirc flow and affect on voids

B. Plausible – bypass valve will affect level / voids but later in the sequence of events

C. Correct: shrink effect

D. Plausible – misconception of level response TCV closure results in shrink

Technical Reference(s): LOT-00-216 p 59 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-216 EO K5.12 (As available)

Question Source: Bank # WTSI 13950

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2010 Monticello

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028	EK3.01
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE : Emergency depressurization

Proposed Question: RO Question # 46

If Drywell temperature cannot be restored and maintained less than 280°F, EOP-3 Primary Containment Control, directs you down a path that can result in RPV-ED.

Which ONE of the following describes the basis for this temperature limit?

A. This is the design temperature of the Containment, which may fail above this temperature.

B. Parameters cannot be maintained on the safe side of the Drywell Spray Initiation Limit Curve, at or above this temperature.

C. Parameters cannot be maintained on the safe side of the RPV Saturation Limit Curve, at or above this temperature.

D. The Heat Capacity Temperature Limit of the Suppression Pool cannot be maintained in the Safe region, if the Drywell is at or above this temperature.

Proposed Answer: A

Explanation (Optional):

A. Correct.

B. Incorrect - Even with DW temperature above 280 °F you can still be on the Safe side of the DWSIL curve.

C. Incorrect - Even with DW temperatures greater than 260 °F you may still be on the Safe region of the RPV Saturation Limit Curve. It depends on what RPV pressure and reference leg temperatures are.

D. Incorrect - The HCTL is based on Torus temperature & Level and RPV pressure (not DW temp.).

Technical Reference(s): EOP Study Guide Volume 4, Section 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-607 EO 3 (As available)

Question Source: Bank # VY LOI 1708

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK3.02	
	Importance Rating	3.7	

Knowledge of the reasons for the following responses as they apply to CONTROL ROOM ABANDONMENT : Turbine trip
Proposed Question: RO Question #47

Given the following:

- A fire has created a need to abandon the Control Room.
- The crew is implementing OPOP-ALTSD-3126, Shutdown Using Alternate Shutdown Methods.

Which of the following describes how and why the Main Turbine is tripped/isolated in this situation?

A. Manual reactor scram from the control room initiates a turbine trip to prevent an uncontrolled RPV cooldown.

B. The turbine is manually tripped from the control room since a fire could cause hot shorts and only the trip from the control room can be assured to work.

C. MSIVs are closed to ensure the turbine is tripped since the relays that actuate this would not be affected by the fire.

D. The turbine is locally tripped after the control room has been evacuated since a fire could cause hot shorts and only the local trip can be assured to work.

Proposed Answer: A

Explanation (Optional):

A. Correct. With the potential for hot shorts, not all steam valves may operate correctly. When the reactor is scrammed then the turbine will trip on low scram air header pressure to prevent a continuos cooldown of the RPV.

B. Incorrect. There are no actions to trip the turbine using turbine controls from the control room in OPOP-ALTSD-3126. Plausible because turbine can be manually tripped from control room and hot shorts could cause spurious operation if it is not addressed right away.

C. Incorrect. MSIVs are closed to maintain inventory, not to ensure that the turbine is tripped. Plausible because closing MSIVs interrupts steam flow to the turbine but any relays and their cabling could be affected by the fire.

D. Incorrect. Tripping the turbine locally on a control room evacuation is logical and plausible but it is not the way the procedure is performed.

Technical Reference(s): OPOP-ALTSD-3126 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-00-612 Objective 1.1 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018	AA1.02
	Importance Rating	3.3	

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : System loads

Proposed Question: RO Question # 48

Given the following:

- The plant is at rated power.
- The running RBCCW pump has tripped.
- The standby RBCCW pump will NOT start either automatically or manually.

Which of the following actions is required for an extended loss of RBCCW flow?

A. Run the Recirculation Pumps to minimum speed to maintain temperatures within limits.

B. Scram the reactor and isolate radwaste due to loss of cooling.

C. Isolate the NRHX from RBCCW and establish alternate cooling water from Service Water.

D. Isolate RBCCW cooling to CRD and RHR pump coolers, and establish alternate cooling from Service Water.

Proposed Answer: D

Explanation (Optional):

A. Incorrect. Recirc pumps are tripped but this is plausible because running them back to minimum speed logically reduces the amount of cooling required.

B. Incorrect. The reactor is scrammed but Radwaste gets alternate cooling, it does not get isolated.

C. Incorrect. NRHX loses cooling on loss of RBCCW but it does not get isolated from RBCCW, and does not receive alternate cooling from service water.

D. Correct. These actions are taken in accordance with OPON-3147-01

Technical Reference(s): OPON-3147-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-603 RO 3g (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031	EA1.13
	Importance Rating	4.3	

Ability to operate and/or monitor the following as they apply to REACTOR LOW WATER LEVEL : Reactor water level control

Proposed Question: RO Question # 49

Given the following:

- The plant has experienced a LOCA with a loss of all high pressure injection.
- Both Recirc pumps have tripped on low-low level.
- All low pressure ECCS pumps are currently running but not injecting.
- As a result, the CRS has ordered an emergency depressurization based on RPV level being < TAF.

What is the preferred RPV level instrument to be used during

the RPV blowdown ___ (1) ___ ?

vessel reflood, ___ (2) ___ ?

A. 1) Compensated RX Level Wide 70 (ERFIS point WIDEM071)
2) Wide Range Level on LR-6-98 on CRP 9-5

B. 1) Wide Range Level on LR-6-98 on CRP 9-5
2) Transient level recorders (LR/PR-2-3-68A/B)

C. 1) Transient level recorders (LR/PR-2-3-68A/B)
2) ECCS Shroud Level Indicators (LI-2-3-91A/B)

D. 1) Compensated RX Level Wide 70 (ERFIS point WIDEM071)
2) ECCS Shroud Level Indicators (LI-2-3-91A/B)

Proposed Answer: D

Explanation (Optional):

A. Incorrect. The wide range recorder would be out of its calibration conditions.

B. Incorrect. The wide range recorder and transient level indicators would be out of their calibration conditions.

C. Incorrect. See explanation for D

D. Correct - During rapid reactor de-pressurization transients, compensated Wide Range or Shroud ERFIS points should be used, WIDE 70 or SHDA or B Compensated Shroud level instruments.
During vessel flood up with ECCS systems, uncompensated shroud level instrument LI-2-3-91A/B (on CRP 9-3) may be used to monitor and control vessel level because they are 'cold' calibrated instruments.

Technical Reference(s): OPAD-0166, Section 6.1.2.B (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-627 EO 1, 2
LOT-00-216, K4.14 (As available)

Question Source: Bank # VY LOR 2225

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001	AA1.07
	Importance Rating	3.1	

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION : Nuclear boiler instrumentation system

Proposed Question: RO Question # 50

Which one of the following describes the expected sudden indication changes, if Jet Pump #1 (Recirc Loop 'B') has failed due to a displaced mixer?

A. Actual Total Core Flow rises.
Recirc Loop 'A' flow rises.

B. Actual Total Core Flow lowers.
Recirc Loop 'A' flow rises.

C. Actual Total Core Flow rises.
Recirc Loops 'A' and 'B' flows both lower.

D. Actual Total Core Flow lowers.
Recirc Loop 'A' flow lowers.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - Total Flow will lower although indicated flow will rise

B. Correct: OPON-3141-01 entry conditions indicate that Recirc loop flow rises in the loop opposite one with a broken jet pump due to less resistance for that intact loop.

C. Incorrect - Total Flow will lower although indicated flow will rise

D. Incorrect - Flow in the 'A' Loop will lower

Technical Reference(s): OPON-3141-01 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-202, Obj 1 (As available)

Question Source: Bank # VY LOR 451

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030	EA2.02
	Importance Rating	3.9	

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool temperature

Proposed Question: RO Question # 51

Using the Heat Capacity Temperature Limit diagram, determine which one of the following sets of parameters VIOLATES the HCTL.

Torus level (ft) / Torus temperature (°F) / RPV pressure (psig)

A. 9 / 180 / 1050

B. 10 / 205 / 480

C. 11 / 220 / 250

D. 12 / 245 / 47

(Note: HCTL graph provided)

Proposed Answer: A

Explanation (Optional):

A. Correct Response- 180 degrees exceeds the 1080 psig curve for 9’ torus level.

B. Incorrect - 205 degrees is below the 500 psig curve for 10’ torus level.

C. Incorrect - 220 degrees is below the 300 psig curve for 11’ torus level.

D. Incorrect - 245 degrees is below the 50 psig curve for 12’ torus level.

Technical Reference(s): EOP-3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: Yes, Provide HCTL graph

Learning Objective: LOT-00-607 K2.06 (As available)

Question Source: Bank # VY 7579

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005	AK3.04
	Importance Rating	3.2	

Knowledge of the reasons for the following responses as they apply to MAIN TURBINE GENERATOR TRIP : Main Generator Trip

Proposed Question: RO Question # 52

Which one of the following describes the setpoint and justification for having a high reactor water level main turbine trip?

A. 177 inches
Provides backup protection for steam line components should the high reactor water level trip of the feed pumps fail.

B. 177 inches
Isolates the turbine to prevent moisture carryover from the reactor.

C. 165 inches
Causes a scram to shut down the reactor, to protect the turbine.

D. 165 inches
Prevents erratic operation of the MHC pressure control system caused by the effect of higher density steam on turbine speed.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - Isolates the turbine to prevent moisture carryover from the reactor.

B. Correct Response

C. Incorrect - 165 inches is the RPV level hi alarm. No scram directly from high reactor water level

D. Incorrect -165 inches is the RPV level hi alarm. Isolates the turbine to prevent moisture carryover from the reactor.

Technical Reference(s): LOT-01-245 (Attach if not previously provided)
OT 3114

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-01-245 K4.18 (As available)

Question Source: Bank # VY LOI 3758
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019	AA2.01
	Importance Rating	3.5	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Instrument air system pressure

Proposed Question: RO Question # 53

Given the following plant conditions:

- With the plant operating at power, a loss of instrument air occurs.
- Instrument air header pressure is 90 psig and lowering.
- Station service air compressors 'A' through 'D' are running loaded.

Which of the following is the correct system and operator response?

Ensure PCV-1…

A. is fully closed when header pressure drops to 85 psig, and manually scram the plant when the first accumulator alarm comes in.

B. is fully open when header pressure drops to 85 psig, and manually scram the plant if control rods begin to drift in.

C. is fully closed at 80 psig and manually scram the plant at <55 psig scram air header pressure.

D. begins to open when header pressure drops to 80 psig, and manually scram the plant when the first accumulator alarm comes in.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - PCV-1 is not fully closed until 80 psig; accumulator alarms are NOT action initiators for this condition.

B. Incorrect - PCV-1 is not fully closed until 80 psig.

C. Correct.

D. Incorrect - Accumulator alarms are NOT action initiators for this condition.

Technical Reference(s): OPON-3146-01; OP 2190 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-279 CRO 1e, 4, 5 (As available)

Question Source: Bank # VY LOR 972
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004	2.1.27
	Importance Rating	3.9	

Conduct of Operations: Knowledge of system purpose and / or function.
Proposed Question: RO Question # 54

The plant has experienced a small break LOCA and the EOPs have been entered.

- Both Emergency Diesel Generators (EDG) are running.
- High Pressure Coolant Injection (HPCI) is operating in pressure control.
- Reactor Core Isolation Cooling (RCIC) is operating in level control.

Subsequently, a loss of DC-1 occurs.

How is the operation of HPCI, RCIC, and the EDGs affected?

A. The HPCI and RCIC high level trip would not function.
The control power to 'A' EDG would be lost.

B. The HPCI high level trip would not function, RCIC would be unaffected.
The control power to 'A' EDG would be lost.

C. The RCIC high level trip would not function, HPCI would be unaffected.
The control power to the 'B' EDG would be lost.

D. The HPCI and RCIC high level trip would not function.
The control power to the 'B' EDG would be lost.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - control power is lost to the 'B' EDG not the 'A'

B. Incorrect - RCIC high level trip is also lost and the 'B' EDG loses control power

C. Incorrect - HPCI high level trip is also lost

D. Correct Response- HPCI and RCIC high level trip would not function. The control power to the 'B' EDG would be lost.

Technical Reference(s): OPON-3159-01 (Attach if not previously provided)

OP-2145

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-263 K3.01, K3.02, K3.03 (As available)

Question Source: Bank # VY LOI 995

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038	2.1.30
	Importance Rating	4.4	

Conduct of Operations: Ability to locate and operate components, including local controls.

Proposed Question: RO Question # 55

- The plant has entered ON 3152, MSL and OFF GAS High Radiation on high SJAE radiation levels.
- The steam packing exhauster is in service.

Control Room personnel were unable to place the Shutdown Iodine Filter in service from the control room.

Which ONE of the following is the correct action for the above conditions?

A. At Turbine Building SJAE ROOM, locally place the Shutdown Iodine Filter in service.

B. At Turbine Building Hallway Rack 27, locally place the Shutdown Iodine Filter in service.

C. Attempt to determine if MSL radiation levels are increasing due to a resin or glycol intrusion into the RPV, hydrogen water chemistry (HWC), or NobleChem.

D. If MSL radiation levels reach ~2.5X normal, stop any power changes until evaluated by Design Engineering.

Proposed Answer: B

Explanation (Optional):

A. Incorrect.

B. Correct. See reference

C. Incorrect.

D. Incorrect.

Technical Reference(s): OPOP-AOG-2150 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000	AK2.04
	Importance Rating	2.5	

Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Breakers, relays, and disconnects

Proposed Question: RO Question # 56

Given the following:

- The Fire Brigade is fighting a fire in the south warehouse.
- The Electric Fire Pump started and immediately tripped on overcurrent.
- The Diesel Fire Pump is running and supplying the fire system with system pressure at 90 psig.
- The overcurrent trip of the Electric Fire Pump breaker is reset.

Which ONE of the following describes the electric and diesel fire pump response?

A. The Electric Fire Pump starts and the Diesel Fire Pump continues to run.

B. The Electric Fire Pump starts and the Diesel Fire Pump stops.

C. The Electric Fire Pump remains in standby and the Diesel Fire Pump continues to run.

D. The Electric Fire Pump remains in standby and the Diesel Fire Pump stops.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Pressure above 85 psig, electric pump will not start.

B. Incorrect -Pressure above 85 psig, electric pump will not start.

C. Correct Response- To secure the diesel driven FP, must, ensure system pressure is above 85 psig and manually take to off. Only the electric auto secures after 7 minutes above 85 psig. When pump reset, pressure above 85 psig so will not auto start and diesel does not auto secure.

D. Incorrect -Diesel FP does not auto secure

Technical Reference(s): OP-2186 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-286 K4.01, K4.02 (As available)

Question Source: Bank # VY LOI 5673

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024	EK1.01
	Importance Rating	4.1	

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL PRESSURE : Drywell integrity: Plant-Specific

Proposed Question: RO Question # 57

In EOP-3, Primary Containment Control, Drywell Sprays are only initiated when Drywell parameters are in the Safe region of the “Drywell Spray Initiation Limit” curve.

What would be the effect of initiating DW Spray while in the **Unsafe** region of the curve?

A. Over pressurization and exceeding Drywell design internal pressure limits.

B. The collapse of the downcomers, due to excessively high dp with DW pressure higher than Torus pressure.

C. Drywell Sprays causing electrical damage to the reactor Recirc Pumps and Drywell RRU's.

D. Containment failure/de-inerting after Drywell Sprays are initiated.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - Over pressurization is not a concern. When drywell sprays are initiated DW pressure will be reduced.

B. Incorrect - Adherence to the DW spray initiation curve prevents the DW to Torus pressure from becoming excessively negative due to evaporative cooling in the DW.

C. Incorrect - It is expected that the DW spray droplets will flash into steam. The basis of the DWSIL Curve is to ensure the ensuing evaporative cooling does not reduce the DW to Torus dp below the capability of the vacuum breaker and the containment pressure does not fall below atmospheric pressure.

D. Correct - The Evaporative Cooling pressure drop is so rapid and effective that it may exceed the capacity of the vacuum breakers to makeup to the Drywell quickly enough to prevent failure due to exceeding its negative pressure limit.

Technical Reference(s): EOP Study Guide section 8 pg. 12-15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-607, STA EO 3 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:
VY Bank 1291

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023	2.4.45
	Importance Rating	4.1	

Emergency Procedures / Plan: Ability to prioritize and interpret the significance of each annunciator or alarm.

Proposed Question: RO Question # 58

The plant is in refuel mode with the following initial conditions:

- Fuel offload is in progress.
- The drywell is open with an air purge of the drywell in progress.

Then the following alarms and indications are received.

- Alarm RX BLDG/REFUEL FLR CH A RAD HI (5-H-1) annunciates
- CRP 9-10 readings are as follows:
  - Refuel Floor East Rad Mon Ch. ‘B’ indicates 85 mR/hr and rising slowly.
  - Refuel Floor West Rad Mon Ch. ‘A’ indicates 110 mR/hr and rising slowly.
  - Rx Bldg Vent Exh North Rad Mon Ch. ‘B’ indicates 10 mR/hr and rising slowly.
  - Rx Bldg Vent Exh South Rad Mon Ch. ‘A’ indicates 12 mR/hr and rising slowly.

Which ONE of the following is the correct priority regarding any operator action and automatic response to the above conditions?

A.
1. Evacuate the reactor building.
2. SBGT will start and Rx Building ventilation will isolate.
3. The drywell air purge lineup will isolate.

B.
1. Evacuate the reactor building.
2. SBGT will start and Rx Building ventilation will isolate.
3. The drywell will remain aligned for air purge.

C.
1. Evacuate the refuel floor only.
2. NO automatic actions will occur until at least one reactor building ventilation exhaust monitor exceeds its high level trip setpoint.

D. 1. Evacuate the refuel floor only.
2. NO automatic actions will occur until both refuel floor monitors exceed their high level trip setpoint.

Proposed Answer: A

Explanation (Optional):

A. Correct: A single refuel floor radiation monitor upscale trip will result in Group 3 isolation. The upscale trip for these monitors is 100 mR/hr. A Group 3 Isolation results in the shutdown of the reactor building ventilation system, starting of both SBGT trains, and the isolation of primary containment purge and exhaust paths.

B. Incorrect: The purge lineup will also isolate.

C. Incorrect: The Group 3 isolation will also occur if a single refuel radiation monitor exceeds its upscale trip.

D. Incorrect: The Group 3 isolation will occur if a single refuel radiation monitor exceeds its upscale trip. In this case the Refuel Floor West Rad Mon Ch. A has exceeded the 100 mR/hr trip setpoint.

Technical Reference(s): LOT-00-272 (Attach if not previously provided)
ARS 5-H-1
OP 1101

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-272 EO k8.06 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 11

55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295017	AK1.02
Importance Rating	3.8	

Knowledge of the operational implications of the following concepts as they apply to HIGH OFF-SITE RELEASE RATE : Protection of the general public

Proposed Question: RO Question # 59

A reactor pressure transient has occurred causing a violation of the high power MCPR safety limit of __(1)__ for two loop operation, and the resulting fuel damage could cause a member of the public to exceed a small fraction of the 10CFR100 site boundary dose limit __(2)__ Rem TEDE .

A. (1) 1.09
(2) 25

B. (1) 1.10
(2) 25

C. (1) 1.09
(2) 5

D. (1) 1.10
(2) 5

Proposed Answer: A

Explanation (Optional):

A. Correct

B. Incorrect. 1.10 is plausible because it is for single loop operation

C. Incorrect. Off-site limit is too low. Limit is 25

D. Incorrect. See B and C

Technical Reference(s): ODCM, TS1.1 (Attach if not previously provided)
10CFR100

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-138, Obj 17, 19 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:
VY Bank 5696

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032	EK2.02
	Importance Rating	3.6	

Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Secondary containment ventilation

Proposed Question: RO Question # 60

Given the following:

- Following a plant trip caused by a low reactor water level the Reactor Building ventilation isolated on a Group III signal.
- RPV water level has been restored to normal.
- While isolated Reactor Building temperatures have risen to 106 F in the reactor building 252 foot level.

What actions are required to restore normal Reactor Building Ventilation?

Place all Group 3 isolation valves in the ____ (1) _____ position.

Ensure containment isolation reset permissive lights (IOPL) are ____ (2) ____ and then reset the Group 3 Isolation.

After re-opening HVAC 9, 10, 11, and 12;

start normal RB HVAC by holding Reactor Bldg ____(3)____ Fan control switch to ON until exhaust and supply fans are running.

A. (1) CLOSE
(2) ON
(3) Exhaust

B. (1) OPEN
(2) OFF
(3) Exhaust

C. (1) CLOSE
(2) OFF
(3) Supply

D. (1) OPEN
(2) ON
(3) Supply

Proposed Answer: A

Explanation (Optional):

A. Correct – Group 3 valves and dampers are placed in CLOSED positions, lights are verified ON, then reset the isolation and hold Reactor Bldg Exh Fan REF-1A(1B) control switch to ON until exhaust and supply fans are running.

B. Incorrect - Group 3 valves and dampers are placed in CLOSED, lights are ON

C. Incorrect - lights are ON, Exhaust fan switch starts fans

D. Incorrect - Group 3 valves and dampers are placed in CLOSED. Exhaust fan switch starts fans

Technical Reference(s): OP 2115
OP 2192 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 4
55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments: VY 2009 Audit 65

Examination Outline Cross-reference:

Level	RO	SRO
Tier #	1	
Group #	2	
K/A #	295013	AK3.01
Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION POOL TEMPERATURE : Suppression pool cooling operation

Proposed Question: RO Question # 61

Given the following conditions:

- Reactor power is 82%.
- HPCI Full Flow Test quarterly Tech. Spec. surveillance is in progress.
- 'B' loop of RHR is in torus cooling.

Which of the below identifies the action(s) required when torus temperature exceeds 90°F?

A. Enter EOP-3 and place 'A' loop of RHR in torus cooling to cool the torus to less than 90°F.

B. Scram the reactor and enter EOP-1 since torus water temperature has exceeded its action point.

C. Terminate the HPCI surveillance regardless of its extent of completion since torus water temperature is greater than 90°F.

D. Declare HPCI inoperable prior to exceeding 100°F torus water temperature due to HPCI oil system potential failure.

Proposed Answer: A

Explanation (Optional):

A. Correct Response -Correct, based on step PC/TT-1 and TT-2

B. Incorrect - The procedural direction for scramming the reactor and entering EOP-1 does not occur until Torus Temperature is 110°F.

C. Incorrect - The HPCI surveillance is required by TS to prove operability. The Surveillance can be completed with little or no rise in torus temperature by placing the 2nd loop of RHR in torus cooling.

D. Incorrect - The temperature in the torus that would be a concern to HPCI operability, due to the inability to cool HPCI lube oil, is 140°F. The basis for the 100°F limitation on Torus Temperature is an operability issue for the Primary Containment pressure suppression function

Technical Reference(s): EOP-3, Entry Conditions and Steps PC/TT-1 and PC/TT-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-607, SRO 2 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295007	AA1.03
	Importance Rating	3.4	

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE : RCIC: Plant-Specific

Proposed Question: RO Question # 62

RCIC is injecting to the reactor vessel with the flow controller in MANUAL at 400 gpm.

Reactor pressure rises from 850 psig to 950 psig.

Which one of the following will be the effect on RCIC, steady state to steady state?

A. RPM will rise, pump flow will rise.

B. RPM will rise, pump flow will remain constant.

C. RPM will remain constant, pump flow will remain constant.

D. RPM will remain constant, pump flow will lower.

Proposed Answer: D

Explanation (Optional):

A. Incorrect -With the controller in manual the speed will be constant. The affect as seen in the plant and the simulator is that any flow rise is negated by the pressure rise.

B. Incorrect -With the controller in manual the speed will be constant. Pump flow may remain constant with rising RPM due to the rising head it must pump to but it would be difficult to determine without values. The affect as seen in the plant and the simulator is that any flow rise is negated by the pressure rise.

C. Incorrect -RPM will be constant but with RPV pressure rising the head it must pump to will rise and pump flow will lower.

D. Correct.

Technical Reference(s): OP 2121; OP 4121 (Attach if not previously provided)
LOT-00-217

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-217, K1.03 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 8
55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295033	EA2.02
	Importance Rating	3.1	

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Equipment operability

Proposed Question: RO Question # 63

Given the following:

- An unisolable steam leak from RCIC has caused the crew to scram the plant.
- Reactor building vent exhaust is 5 mr/hr.
- The CRS has ordered the BOP to restart reactor building HVAC.

Which ONE of the following describes the basis or correct response for this action?

A. RB HVAC should NOT be started with a leak and RB vent radiation present.

B. Establishes normal RB ventilation flow while assuring a monitored and elevated release.

C. Assures that the RB HVAC is able to maintain a positive pressure in the reactor building.

D. Assures that any radioactive discharge is contained in the primary and secondary containments.

Proposed Answer: B

Explanation (Optional):

A. Incorrect -RB ventilation is desirable under certain rad conditions such as these.

B. Correct Response- EOP-4 study guide, section 10, pg. 5 of 8, Restating RB ventilation ensures continued personnel access for operation of plant equipment essential for responding to plant emergencies or transients that could degrade into plant emergencies.

C. Incorrect -RB HVAC and SBGT are designed to maintain a negative pressure inside the secondary containment.

D. Incorrect -RB HVAC will not assist maintaining any release inside primary & secondary containment.

Technical Reference(s): EOP-4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-611 A3 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 2.4.34	
	Importance Rating	4.2	

Emergency Procedures / Plan: Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

Proposed Question: RO Question # 64

Given the following:

- The control room has been abandoned.
- Shutdown cooling has been established.
- Drywell temperature rises to 270°F as indicated on TI-16-19-42A.
- Drywell pressure is 2.6 PSIG.

IAW OPOP·ALTSD-3126, SHUTDOWNUSING ALTERNATE SHUTDOWN METHODS, the operators will:

A. Take no action at this time.
Action is not required by the procedure until Drywell temperature exceeds 325°F.

B. Spray the Drywell by placing RHR 'A' in torus cooling and manually opening RHR 26A and RHR 31A.

C. Take no action.
Cooling the reactor with shutdown cooling will remove the heat source and ambient losses will cool the Drywell.

D. Spray the Drywell using RHR 'A' by securing shutdown cooling and opening RHR 26A and RHR 31A from the Alternate Shutdown Panel.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - Action is required at 260°F

B. Correct - IAW OPOP·ALTSD-3126, App A

C. Incorrect - Action is required at 260°F

D. Incorrect - RHR 26A and RHR 31 A cannot be operated from the Alternate Shutdown Panel

Technical Reference(s): OPOP-ALTSD-3126 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-601 EO 8 (As available)

Question Source: Bank # WTSI 3771

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 Vermont Yankee

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295020	2.4.1
	Importance Rating	4.6	

Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: RO Question # 65

Given the following conditions:

- The plant was operating at 100% power.
- A Feed Regulating Valve malfunction occurred and OT 3113 was entered.
- During the transient, RPV level lowered to 78 inches.
- All automatic actions occur as designed, with no other equipment issues occurring.

Subsequently,

- FRV malfunction has been corrected and reactor level has been restored to 150 inches.
- HPCI and RCIC setpoints have been dialed down and are running on minimum flow.
- Feedwater and Condensate are operating with 1 RFP and 1 Condensate pump.

Which ONE of the following is the correct long term response?

A. Establish torus cooling and re-open the MSIVs.

B. Establish torus cooling and shift HPCI to the pressure control mode.

C. Establish torus cooling and shift RCIC to the pressure control mode.

D. Commence cooldown using SRVs at <100 F per hour.

Proposed Answer: A

Explanation (Optional):

A. Correct Response- Basis document OT-3100, Any time the main condenser is available as a heat sink and it is safe to do so, the operator is instructed to use it rather than the torus as a heat sink. This will minimize the heat load on the primary containment. The

S/-8 provides the conditions under which the MSIVs may be reopened.

B. Incorrect - This would add more heat to the torus which is never the preferred action

C. Incorrect - This would add more heat to the torus which is never the preferred action

D. Incorrect - This would add more heat to the torus which is never the preferred action

Technical Reference(s): EOP Basis OT-3100; EOP-1, OT-3100 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-600 CRO A2, A3 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.1	
	Importance Rating	3.8	

Knowledge of conduct of operations requirements.
Proposed Question: RO Question # 66

Which ONE of the following identifies two activities, both of which are listed as on Attachment 9.9 of EN-OP-115, Conduct of Operations, as being approved for two-handed operations?

A. Starting a RWCU pump while adjusting CU-74 (Demineralizer Bypass Valve); AND While paralleling a DG to the 4KV Bus, Adjust DG output voltage and speed until the synchroscope rotates slowly in the FAST direction.

B. Starting a RWCU pump while adjusting CU-74 (Demineralizer Bypass Valve); AND Closing the first generator output breaker and picking up load on the Main Generator.

C. Realigning RCIC components after an **inadvertent** actuation; AND Closing the first generator output breaker and picking up load on the Main Generator.

D. Realigning RCIC components after an **inadvertent** actuation; AND While paralleling a DG to the 4KV Bus, Adjust DG output voltage and speed until the synchroscope rotates slowly in the FAST direction.

Proposed Answer: B

Explanation (Optional):

A. **Incorrect:** 1st part correct, 2nd part wrong. This is incorrect because only the first activity is listed. This is plausible because the second activity is a routine activity that requires precise subsequent actions, and the operator may incorrectly believe that two-handed actions are approved.
B. **Correct:** 1st part correct, 2nd part correct. In accordance with Attachment 9.9 of EN-OP-115 (p85; Rev 14), both activities are specifically listed among the nine acceptable activities.
C. **Incorrect:** 1st part wrong, 2nd part correct. This is incorrect because only the second activity is listed. This is plausible because the first activity is a routine activity, and the

operator may incorrectly believe that two-handed actions are approved.

D. **Incorrect:** 1st part wrong, 2nd part wrong. See B and C.

Technical Reference(s): Attachment 9.9 of EN-OP-115 (Attach if not previously provided)
(p85; Rev 14)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G1 2.1.25	
	Importance Rating	3.9	

Conduct of Operations: Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: RO Question # 67

The control room has been evacuated due to fire in the control room.
Shutdown from outside the control room is in progress.
A cooldown with SRVs has been initiated.
The following data is taken at the indicated times.

TIME (HH:MM)	RX PRESSURE (psig)
00:30	760
00:40	680
00:50	600
01:00	520
01:10	440
01:20	360
01:30	280

Which ONE of the following groups of statements accurately describes the status of the cooldown rate?

A. The cooldown rate over the last hour averaged less than 90°F per hour.

The cooldown rate for the last 10 minutes averaged less than 90°F per hour.

B. The cooldown rate over the last hour averaged greater than 90°F per hour.

The cooldown rate for the last 10 minutes averaged greater than 90°F per hour.

C. The cooldown rate over the last hour averaged less than 90°F per hour.

The cooldown rate for the last 10 minutes averaged greater than 90°F per hour.

D. The cooldown rate over the last hour averaged greater than 90°F per hour.

The cooldown rate for the last 10 minutes averaged less than 90°F per hour.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. No other time period exceeded rate specified in question

B. Correct. Per steam table (must convert psia to psig)
00:30 to 01:30 = 101 °F >90
01:20 to 01:30 = 23 °F X 6 =138, >90
Per OPOP-ALTSD-3126 a 90 F/hour cooldown rate is prescribed.

00:00	1000 psig - 544 Deg. F.	00:50	600 psig - 486
00:10	920 psig - 535	01:00	520 psig - 471
00:20	840 psig - 524	01:10	440 psig - 454
00:30	760 psig - 512	01:20	360 psig - 434
00:40	680 psig - 500	01:30	280 psig - 411

C. Incorrect. No other time period exceeded rate specified in question

D. Incorrect. No other time period exceeded rate specified in question

Technical Reference(s): OPOP-ALTSD-3126 (Attach if not previously provided)
Steam Tables

Proposed References to be provided to applicants during examination: Steam Tables
Steam tables

Learning Objective: LOT-00-612 TO 9 (As available)

Question Source: Bank # WTSI 580
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2007 Vermont Yankee

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

Principles of heat transfer, thermodynamics and fluid mechanics.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2 2.2.1	
	Importance Rating	4.5	

Equipment Control: Ability to perform pre-startup procedures for the facility, including operating those controls associated with plant equipment that could affect reactivity.

Proposed Question: RO Question # 68

Given the following:

- The plant has been in a maintenance outage.
- Plant conditions have been established to start 'A' Recirculation Pump in preparation for reactor startup.
- The Operator starts the 'A' Recirculation pump and observes normal panel indication for start of the MG Set drive motor.

Shortly after, the following alarms are received:

- 4-A-1 (MG SET A GEN LOCKOUT)
- 4-A-2 (MG SET A GEN AUX LOCKOUT)
- 4-A-7 (MG SET A SEQ INCOM)
- 4-A-5 (MG SET A DRIVE MOTOR TRIP)

Which of the following caused the trip of the 'A' Recirculation Pump?

A. Recirc MG Set lube oil pressure was 35 psig.

B. Recirc MG Set lube oil temperature was 195°F.

C. Field breaker did not close within 15 seconds of MG drive motor breaker closure.

D. The discharge valve was <90% open after 5 minutes.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: This is below the drive motor trip of 40 psig for a 6 second TD but would not cause an incomplete sequence

B. Incorrect: Recirc MG Set lube oil temperature will cause a drive motor breaker trip if above 210°F

C. **CORRECT:** Field breaker did not close within 15 seconds of MG drive motor breaker closure will result in the trip. The incomplete sequence brings in the LOCKOUT alarms and Trips the drive motor

D. Incorrect: The drive motor breaker trip is valid but would not cause the MG SET A SEC INCOM

Technical Reference(s): OP-2110
ARS 4-A-7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-202 (As available)

Question Source: Bank # WTSI 12980
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Monticello

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

10 CFR Part 55 Content: 55.41 7
55.43

Design, components, and function of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	G2 2.2.39	
	Importance Rating	3.9	

Equipment Control: Knowledge of less than or equal to one hour technical specification action statements for systems.

Proposed Question: RO Question # 69

The plant was shutdown four days ago for a refueling outage, and maintenance with the potential to drain the reactor vessel (OPDRV) is in progress.

An Auxiliary Operator contacts the Control Room and informs them that someone has blocked open both reactor building airlock doors.

Which one of the following actions is required?

A. Within four hours verify one airlock door closed OR stop maintenance with the potential to drain the reactor vessel (OPDRV).

B. Immediately stop maintenance with the potential to drain the reactor vessel (OPDRV) .

C. Within four hours verify one airlock door closed OR stop any refueling activities on the Refuel Floor.
Maintenance with the potential to drain the reactor vessel (OPDRV) may continue.

D. Immediately stop any refueling activities on the Refuel Floor and initiate action to close at least one air lock door.
Maintenance with the potential to drain the reactor vessel (OPDRV) may continue.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - immediately

B. Correct - With Secondary Containment inoperable during movement of recently irradiated fuel assemblies (24 hours) in the secondary containment and during (OPDRV) - Initiate actions to suspend OPDRVs

C. Incorrect - immediately and maintenance with the potential to drain the reactor vessel (OPDRV) must be stopped

D. Incorrect - immediately and maintenance with the potential to drain the reactor vessel (OPDRV) must be stopped

Technical Reference(s): TS 3.7.C.4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-308 EO 1 (As available)

Question Source: Bank # WTSI 3659

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2009 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.13	
	Importance Rating	3.4	

Radiation Control: 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: RO Question # 70

Which conditions below requires the evacuation of the Refuel Floor?

1. Doubling of SRM counts.
2. High Activity alarm from any airborne activity monitor on the Reactor Building 345' elevation.
3. A dropped fuel assembly.
4. Unanticipated rise in reactor cavity level.
5. Unanticipated high radiation alarm on the refuel floor area radiation monitor.

A. 1, 2 and 4

B. 2, 3 and 5

C. 3, 4 and 5

D. 1, 2 and 5

Proposed Answer: B

Explanation (Optional):

A. Incorrect - 1 and 4 do not require evacuation of the Refuel Floor

B. Correct

C. Incorrect - 4 does not require evacuation of the Refuel Floor

D. Incorrect - 1 does not require evacuation of the Refuel Floor

Technical Reference(s): OP 1101 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-601 EO3 (As available)

Question Source: Bank # WTSI 3762
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Vermont Yankee

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

10 CFR Part 55 Content: 55.41 10
55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.7	
	Importance Rating	3.5	

Radiation Control: Ability to comply with radiation work permit requirements during normal or abnormal conditions.

Proposed Question: RO Question # 71

Following a failure of a Traversing In-Core Probe (TIP) an operator is being sent into the TIP Room to verify the detector position.

Which one of the following describes the type of Radiation Work Permit (RWP) required and its associated requirements in accordance with EN-RP-105, "Radiation Work Permits" ?

A. A Specific RWP with radiological brief given by the Shift Manager.

B. A Specific RWP with documented pre-job brief given by Radiation Protection.

C. Entry may occur under the General RWP for the Reactor Building provided a radiological brief is given by the Shift Manager.

D. Entry may occur under the General RWP for the Reactor Building provided a documented pre-job brief is given by Radiation Protection.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - A Specific RWP would be used because of the potential for high rad levels and the unpredictability of the job, a documented pre-job brief given by Radiation Protection is required

B. Correct - A Specific RWP would be used because of the potential for high rad levels and the unpredictability of the job, a documented pre-job brief given by Radiation Protection is required

C. Incorrect - A General RWP that provides radiological controls of operations, inspections, and maintenance in radiologically controlled areas where radiological conditions are static or changes are anticipated or predictable. A documented pre-job brief given by

Radiation Protection is required

D. Incorrect - A General RWP that provides radiological controls of operations, inspections, and maintenance in radiologically controlled areas where radiological conditions are static or changes are anticipated or predictable

Technical Reference(s): EN-RP-105 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-404 K2.3.17 (As available)

Question Source: Bank # WTSI 3764
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Vermont Yankee

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

10 CFR Part 55 Content: 55.41 12
55.43

Radiological safety principles and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	G4 2.4.2	
	Importance Rating	4.5	

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions.

Proposed Question: RO Question # 72

The plant was operating at 100% power when the operators scrammed the reactor due to unexplained rising drywell pressure.

The reactor was scrammed but power is at 4% due to 15 stuck control rods.

Drywell cooling was lost and has not been restored.

Current plant conditions are:

- Reactor Power at 4%.
- Reactor Pressure 500 psig.
- Drywell Temperature 140°F and rising.
- Drywell Pressure 2.5 psig and rising.
- Reactor water level initially dropped to 80 inches but has recovered to 130 inches.

In addition to EOP-1;

(1) What EOP entry conditions which were met due to this event?

(2) What is the reason for the automatic plant response of the RHR, Core Spray, and 4160 VAC systems?

A. (1) EOP-2 was entered as due to reactor water level low.
EOP-3 was entered due to drywell temperature high.

(2) Low reactor water level initiated the start all RHR and Core Spray pumps and the EDGs.

B. (1) EOP-2 was entered as directed from EOP-1 due to an ATWS condition.
EOP-3 was entered due to drywell pressure high.

(2) High Drywell Pressure initiated the start all RHR and Core Spray pumps and EDGs.

C. (1) EOP-2 was entered as directed from EOP-1 due to reactor water level low. EOP-3 was entered due to drywell pressure high.

(2) Low reactor water level initiated the start all RHR and Core Spray pumps and the EDGs.

D. (1) EOP-2 was entered as directed from EOP-1 due to an ATWS condition. EOP-3 was entered due to drywell temperature high.

(2) High Drywell Pressure initiated the start all RHR and Core Spray pumps and the EDGs.

Proposed Answer: B

Explanation (Optional):

A. Incorrect. EOP-2 is not entered due to reactor water level low, EOP was not entered due to DW temp high (not high enough). Hi DW press. initiated equipment starts

B. Correct. Per EOPs and equipment auto start signals

C. Incorrect. EOP-2 is not entered due to reactor water level low, low water level did not initiate equipment starts

D. Incorrect. EOP was not entered due to DW temp high (not high enough)

Technical Reference(s): EOP-1, EOP-2, EOP-3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-607, Control EO-3 (As available)

Question Source: Bank # WTSI 698

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2007 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.13	
	Importance Rating	4.0	

Knowledge of crew roles and responsibilities during EOP usage.
Proposed Question: RO Question # 73

Given the following:

- The plant was operating at 100% power.
- An event occurred where the reactor should have scrammed but did NOT.
- The crew has entered the EOPs.
- Power remains above 2%.

Assuming that the actions are needed by plant conditions, which ONE of the following identifies RO action(s) that DO(ES) NOT require direction from the CRS before they can be performed?

A. Initiate ARI/RPT and then trip the Recirc Drive Motors, ONLY.

B. Depress the manual SCRAM push buttons; AND

Maintain a reactor water level band of 127 to 177 inches, ONLY.

C. Initiate ARI/RPT and then trip the Recirc Drive Motors; AND

Maintain a reactor water level band of 127 to 177 inches, ONLY.

D. Depress the manual SCRAM push buttons;

Initiate ARI/RPT and then trip the Recirc Drive Motors; AND

Maintain a reactor water level band of 127 to 177 inches.

Proposed Answer: D

Explanation (Optional):

A. **Incorrect:** OPAD-0166 (p5; Rev 00) states that the RO can do this without CRS input if

needed.

B. **Incorrect:** OPAD-0166 (p5; Rev 00) states that upon entering the Emergency Operating Procedures and/or OT 3100 SCRAM Procedure reactor water level band shall be 127 inches to 177 inches unless otherwise directed.

C. **Incorrect:** 1st part can be done without CRS input (See A).

D. **Correct:** OPAD-0166 (p5; Rev 00) states that the RO can do all parts without CRS input if needed.

Technical Reference(s): OPAD-0166 (p5; Rev 00) (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: LOT-03-400 RO1 (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: NA

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	G3 2.3.11	
	Importance Rating	3.8	

Radiation Control: Ability to control radiation releases.

Proposed Question: RO Question # 74

What EOP action(s) are taken to mitigate the consequences of unisolable leakage from a primary system into multiple areas of secondary containment that may pose a direct AND immediate threat to secondary containment integrity?

A. Normal Reactor Shutdown ONLY.

B. Reactor Scram ONLY.

C. Reactor Scram AND Cooldown <100°F/hr.

D. Reactor Scram and Emergency Depressurize the RPV.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - Correct for non-primary system, stem states immediate threat to the Secondary Containment

B. Incorrect - Does not take pressure off, not a full strategy

C. Incorrect -single parameter in excess of max safe requires this action to prevent spreading to more than one area

D. Correct - If any parameter exceeds its Table O value in two areas, then the problem of concern (temperature, radiation, or water level) is not under control and action must be taken to protect the secondary containment. Exceeding the Table O limits in more than one location is indicative of a widespread and worsening condition which could affect secondary containment integrity. Scram and E-Depress is required

Technical Reference(s): EOP-4, OPPP-07018 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-611, EO-4 (As available)

Question Source: Bank # WTSI 3661
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2009 Pilgrim

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

10 CFR Part 55 Content: 55.41 10
55.43
Administrative, normal, abnormal, and emergency operating procedures for the facility.
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	G1 2.1.45	
	Importance Rating	4.3	

Conduct of Operations: Ability to identify and interpret diverse indications to validate the response of another indicator.

Proposed Question: COMMON 75

During a power ascension all APRMS indicate approximately 80%.

Which ONE of the following plant parameters is inconsistent with this value and indicates that the APRMs may have been mis-calibrated?

A. 3D-Monicore indicates that core thermal power is 1525 MWt.

B. Total Steam Flow as indicated on CRP 9-5 is 6.4 Mlbm/hr.

C. Turbine 1st Stage Pressure is 124 psig.

D. Main Generator Output is 504 MWe.

Proposed Answer: C

Explanation (Optional):

A: Incorrect - The APRM system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of design power (1912 MWt). 80% X 1912 = 1530. Therefore the APRMs are consistent with the core thermal power calculation

B: Incorrect: Rated steam flow is 8 Mlbm/hr. 80% X 8 Mlbm/hr = 6.4 Mlbm/hr. Therefore the steam flow is consistent, especially since the APRMs are always set to read higher than actual power

C: Correct: Turbine first stage pressure is normally ~830 psig at 100% and ~650 PSIG at 80%. Either the 1st stage pressure is reading too low or the APRMs are reading too

high. This answer correlates to the turbine stop and control valve fast closure bypass which is at 124 PSIG which the operators should know in maneuvering the plant that it comes in at ~20% reactor power. This is the setpoint of the 5-K-8 alarm.

D: Incorrect: Per the relationship of MWt versus Mwe for plant efficiency, MWe should be ~ 33% of MWT. 1912 MWT=650 MWE. At 80% MWT>1530 MWT X 33%= 504 MWE. Therefore the MWe is also consistent with the APRMs

Technical Reference(s): None Specific (Attach if not previously provided)

(Simulator)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X

Modified Bank # (Note changes or attach parent)

New

Question History: 2011 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes, effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295023 AA2.03	
	Importance Rating		3.8

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS : Airborne contamination levels

Proposed Question: SRO Question # 76

While refueling an irradiated fuel bundle has been dropped and OPON-3153, Excessive Radiation Levels, has been entered.

- Radiation Protection (RP) has been directed to obtain air samples throughout the station due to rising ARM and CAM readings.
- RP now reports that air sample results for the Turbine Building, 272 foot elevation, indicate a current airborne contamination level of 1.2 mrem/hr.

IAW OPON-3153, which one of the following is required?

Direct:

A. That Control Room HVAC Recirc Mode be shifted to EMER and Turbine Building Ventilation be secured to limit any offsite release.

B. That Control Room HVAC Recirc Mode be shifted to EMER and and verify that Turbine Building HVAC is in service to provide an elevated release point.

C. All control room personnel to don SCBAs until the control room atmosphere can be verified to be within limits and Turbine Building Ventilation be secured to limit any offsite release.

D. All control room personnel to don SCBAs until the control room atmosphere can be verified to be within limits and verify that Turbine Building HVAC is in service to provide an elevated release point.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: IAW OPON-3153-01, Turbine Building Ventilation is placed in service if shutdown. Plausible if the candidate believes that this action will reduce the offsite release.

B. Correct: OPON-3153-01, step 3.5, directs that if airborne contamination levels are greater than 0.75 mrem/hr then Turbine Building HVAC is restarted if shutdown and Control Room HVAC is placed in Recirc mode by placing the select switch to EMER.

C. Incorrect: There is no direction to don SCBAs in the control room for this event. Plausible in that SCBAs are donned if Toxic Gas is indicated (see OP 2192, page 5). Additionally Turbine Building Ventilation is placed in service if shutdown.

D. Incorrect: There is no direction to don SCBAs in the control room for this event. Plausible in that SCBAs are donned if Toxic Gas is indicated (see OP 2192, page 5).

Technical Reference(s): LOT-00-611, EOP-4 lesson plan, page 26. (Attach if not previously provided)
OPON-3153, step 3.5

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #
Modified Bank #
New X

Question History: Last NRC Exam: Not Used

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

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295018	AA2.05
	Importance Rating		2.9

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : System pressure

Proposed Question: SRO Question # 77

Given the following:

- A SW break has occurred in the reactor building.
- SW header pressure falls to 48 psig.
- There are no other indications at this time.

One minute after the leak and subsequent pressure drop, which of the following loads will be isolated and what procedure should be entered to mitigate the consequences of the above conditions?

A. RBCCW Heat Exchangers and Steam Tunnel RRUs.
ON 3158, RB Hi area temperature/water level.

B. RBCCW Heat Exchangers and Steam Tunnel RRUs.
ON 3148, Loss of Service Water.

C. RR MG Set Lube Oil Coolers and Generator Hydrogen Coolers.
ON 3148, Loss of Service Water.

D. RR MG Set Lube Oil Coolers and Generator Hydrogen Coolers.
ON 3158, RB Hi area temperature/water level.

Proposed Answer: C

Explanation (Optional):

A. Incorrect -The wrong SW loads are listed and the wrong procedure. There are no indications for high RB water levels.

B. Incorrect -The wrong SW loads are listed. There is an entry condition for ON 3148 with SW header pressure low.

C. Correct Response: These service water loads are isolated when SW pressure falls to <50# for >27 sec. There is an entry condition for ON 3148 with SW header pressure low.

D. Incorrect -These service water loads are isolated when SW pressure falls to <50# for >27 sec. There are no indications for high RB water levels.

Technical Reference(s): OPOP-SW-2181; ON 3148 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-276 CRO 1 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	600000	AA2.03
	Importance Rating		3.2

Ability to determine and interpret the following as they apply to PLANT FIRE ON SITE: Fire alarm

Proposed Question: SRO Question # 78

Given the following:

- With the plant operating at 50% RTP, some solvent that is improperly stored in the Service Water Pump Room ignites.
- Control room fire protection panel, CP-115-3, Zone 15- Intake structure, is in alarm.
- The Control Room is notified of this condition, at 1230 hours, by a contractor that is working in the area.
- The Fire Brigade Leader is dispatched to investigate.
- At 1240 hours, the FBL informs the Control Room that the fire has been extinguished but that the Service Water Pump 'B' power supply cabling has been damaged due to its proximity to the burning solvent.

What (if any) is the required Emergency Classification for this event?

(Note: EALs provided)

A. Unusual Event.

B. Alert.

C. Site Area Emergency.

D. No Classification required, the fire was extinguished in less than 15 minutes.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Safe suhtdown equipment damage took place. If there was only a fire lasting ten minutes then this would be correct.

B. Correct Response - Fire or explosion resulting in visible damage to any Table H-1 area containing safety systems or components or Control Room indication of degraded

performance of safety systems

C. Incorrect: This would be a misinterpretation of the EAL's.

D. Incorrect: The fire was extinguished in time but vital equipment damage took place.

Technical Reference(s): AP-3125, App. A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EALs

Learning Objective: LOT-00-900, 2.4.53 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference: Level RO SRO

Tier #	1
Group #	1
K/A #	295024 2.4.4
Importance Rating	4.7

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters that are entry-level conditions for emergency and abnormal operating procedures.

Proposed Question: SRO Question # 79

The plant is operating at 100% power when the OATC reports:

- Torus water level has dropped from 11.0 ft to 10.81 ft
- Drywell to Torus D/P has dropped from 2.0 psid to 1.8 psid
- Torus pressure has remained constant at 0.0 psig.
- You have confirmed the indications on ERFIS.

Which one of the following actions should the CRS direct?

A. Enter EOP-3, Primary Containment Control, and immediately makeup nitrogen to the Drywell restore D/P.

B. Enter EOP-3, Primary Containment Control, and immediately makeup water to the Torus to restore Torus level.

C. Enter ON-3108, Loss of Primary Containment Integrity, and dispatch Operators to determine why RBCCW temperature has risen.

D. Enter ON-3108, Loss of Primary Containment Integrity, and immediately initiate an orderly shutdown per OP 0105, Reactor Operations.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - EOP-3 Entry conditions do not exist, therefore EOP-3 Entry is not required.

B. Incorrect - EOP-3 Entry conditions do not exist, therefore EOP-3 Entry is not required

C. Incorrect - ON-3108 symptoms exist (Low D/P) warranting entry. A RBCCW temperature rise will cause the reverse of the indications based on higher DW temperatures caused by less heat removal by the DW cooling units.

D. Correct - ON-3108 symptoms exist (Low D/P) warranting entry. The ON states if a loss of primary containment integrity has been verified: While continuing to attempt restoration of primary containment integrity, immediately initiate an orderly shutdown

per OP 0105, Reactor Operations

Technical Reference(s): ARS 21003, 5-G-2, (Attach if not previously provided)
ON 3108, pg 4 of 7

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X
Modified Bank #
New

Question History: 2009 Last NRC Exam: Vermont Yankee

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

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:

Tier #		1
Group #		2
K/A #	295017	2.4.31
Importance Rating		4.1

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: SRO Question # 80

The plant is operating at 100% power when main steam line radiation levels start trending up.

At 1.5 times NFPB (normal full power background) alarm 3-F-1, Mn Stm Ln Rad Hi/Dwnscl alarmed.

Shortly after AEOG Rad Hi Hi alarm (3-G-1) actuated.

Stack 3 Rad monitor indicates normal and steady.

Which procedures should be entered for the above conditions?

A. ON 3152, MSL and Off-gas high radiation and EOP-1, RPV Control.

B. OPON-3153-01, Excessive Radiation Levels and EOP-1, RPV Control.

C. ON 3152, MSL and Off-gas high radiation and EOP-4, Secondary Containment and Radioactivity Release Control.

D. OPON-3153-01, Excessive Radiation Levels and EOP-4, Secondary Containment and Radioactivity Release Control.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – This would be a correct entry condition for ON 3152. There are no EOP-1 entry conditions here.

B. Incorrect – This would be an outdated entry condition for OPON-3153-01 but not EOP-1 as there are no entry conditions.

C. Correct Response - This would be correct entry conditions for ON 3152 along with EOP-4 for the AEOG Rad Hi Hi alarm (3-G-1)

D. Incorrect - This would be an outdated entry condition for OPON-3153-01 but EOP-4 for the AEOG Rad Hi Hi alarm (3-G-1) is correct.

Technical Reference(s): EOP-4 and OPON-3153-01 entry conditions, 3-G-1 redirect actions (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LOT-00-216 k4.04, k5.07 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 8

55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295028 2.4.31	
	Importance Rating		4.1

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures. (High Drywell Temperature)

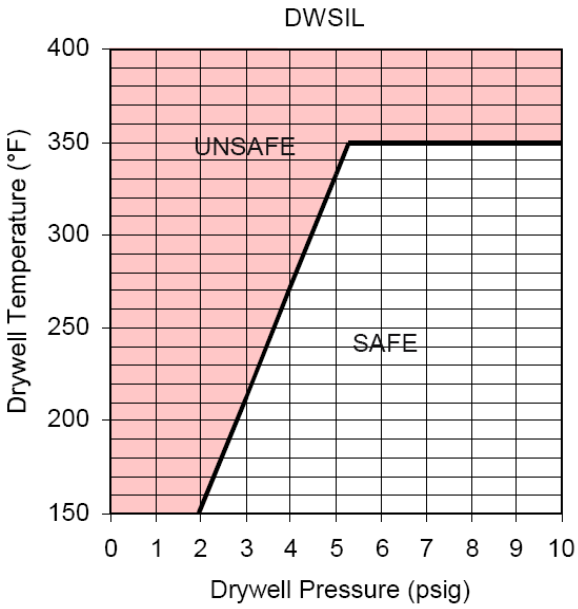
Proposed Question: SRO Question # 81

A LOCA is in progress with the following conditions:

- RPV water level is being maintained in the normal range with feedwater.
- Torus level is 12 feet and rising slowly
- Drywell temperature is 290°F and rising slowly.
- Both recirc pumps and all RRUs are tripped
- Drywell and torus Sprays are available but are not yet in service.
- Drywell pressure is 4.5 psig and rising slowly.

Which one of the following is correct regarding the actions that are to be taken to control drywell temperature per the EOP-3, Primary Containment Control?

Immediately …..



A. enter EOP-5, RPV-ED, to lower drywell temperature.
Do NOT initiate drywell sprays.

B. enter EOP-5, RPV-ED to lower drywell temperature.
Then initiate drywell sprays IAW EOP-3, Primary Containment Control.

C. spray the drywell to lower drywell temperature IAW EOP-3, Primary Containment Control.
Enter EOP-5, RPV-ED, only if sprays do not reduce drywell temperature.

D. spray the drywell to lower drywell temperature IAW EOP-3, Primary Containment Control.
Then, immediately enter EOP-5, RPV-ED, regardless of drywell temperature response.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: An immediate RPV-ED is not required. Plausible in that temperature is above the value associated with the RPV ED step. Additionally the decision not to spray the drywell is also plausible if the candidate does not understand the meaning of a 'Before' step. The direction to spray the drywell is 'Before' the drywell temperature reaches 280 °F.

B. Incorrect: An immediate RPV-ED is not required. Plausible in that temperature is above the value associated with the RPV ED step.

C. Correct: Drywell and temperature and pressure are within the DSIL curve. Although drywell temperature is above the value requiring a blow down, the associated step directs that when temperature 'cannot be restored and maintained' below 280 °F, only then is a blow down performed. As discussed in the EOP study guide, the step allows the operator to attempt to restore the drywell temperature below 280°F because the design temperature limit is on the drywell wall which will remain significantly cooler than the drywell atmosphere. It is not expected to reach 280°F on the drywell wall for some time after exceeding 280°F during the worst case postulated steam leak in the drywell. If after attempting drywell sprays, drywell temperature cannot be restored below 280°F, then RPV-ED is required.

D. Incorrect: An RPV-ED is not immediately required after spraying the drywell. Plausible in that temperature is above the value associated with the RPV ED step.

Technical Reference(s): EOP-3, Primary Containment Control (Attach if not previously provided)
EOP Study Guide, EOP-3 PC Control, Page 17 of 43

Proposed References to be provided to applicants during examination: None

Learning Objective: Lesson Plan LOT-00-607, SRO EO # 3 (As available)

Question Source: Bank # WTS # 11851

Modified Bank #

New

Question History: Last NRC Exam: NMP2 2010

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295037	EA2.01
	Importance Rating		4.3

Ability to determine and/or interpret the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : Reactor power

Proposed Question: SRO Question # 82

Given the following:

- The reactor was operating at 100% power.
- A PCIS Group 1 isolation occurred on low main condenser vacuum.
- All control rods failed to insert.
- Reactor power lowers and stabilizes at 45%.
- 4 SRV's are open on overpressure protection.
- EOP-2 has just been entered.
- Vessel level is currently 86 inches and lowering.
- Torus water temperature is 112°F and rising.

For the above conditions, the operator should terminate and prevent all injection for level-power control at what time?

A. Immediately.

B. Immediately after running back recirc flow.

C. Only after Torus temperature exceeds 110°F, and power remains above 2%.

D. After directing OE 3107 Appendix P, when Torus temperature exceeds 160°F and power remains above 2%.

Proposed Answer: A

Explanation (Optional):

A. Correct Response-Prx is >2%, torus water temperature is 110 F

B. Incorrect -Pumps are tripped ARI/RPT on ATWS immediate

C. Incorrect – This has already occurred so the direction from the override in the EOP moves to immediately.
D. Incorrect - MSIV's are shut based on the stem of the question, this indicates the step in EOP-2 would be N/A. Also the main condenser should be determined unavailable with the low main condeser vacuum Group I isolation.

Technical Reference(s): EOP-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-610, CRS 2.4.18, 2.4.22 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295012	AA2.01
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Drywell temperature

Proposed Question: SRO Question # 83

Given the following:

- The Control Room has been evacuated and a shutdown from outside the control room is in progress.
- Drywell air space temperature is 307°F and is rising at 1°F per minute.

Which ONE of the following describes the proper operator response?

A. Once it is determined that Drywell air space temperature cannot be maintained below 325°F (but prior to reaching 325°F), the operators should open SRV-71A and -71B and inject to the reactor with RHR when RPV pressure lowers below 280 psig.

B. Once it is determined that Drywell air space temperature cannot be maintained below 325°F (but prior to reaching 325°F), the operators should place RHR 'A' in torus spray and initiate drywell sprays (providing the Drywell Spray Initiation Curve is satisfied).

C. When Drywell air space temperature exceeds 325°F, the operators should open SRV-71A and -71B and inject to the reactor with RHR when RPV pressure lowers below 280 psig.

D. When Drywell air space temperature exceeds 325°F, the operators should place RHR 'A' in torus spray and initiate drywell sprays (providing the Drywell Spray Initiation Curve is satisfied).

Proposed Answer: C

Explanation (Optional):

A. Incorrect -The procedure states, "if Drywell air space temperature exceeds 325 deg F…" and does NOT offer the latitude to take action based upon prediction.

B. Incorrect -The procedure states, "if Drywell air space temperature exceeds 325 deg F…" and does NOT offer the latitude to take action based upon prediction; AND, the procedure directs opening of the SRVs and injecting with RHR when below 280 psig.

C. Correct Response - Actions are specified in OPOP-ALTSD-3126 Att 1/2.

D. Incorrect - The procedure directs opening of the SRVs and injecting with RHR when below 280 psig.

Technical Reference(s): OPOP-ALTSD-3126 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-612 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295035 2.4.21	
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (Secondary Containment High Differential Pressure)

Proposed Question: SRO Question # 84

The plant is at 100% power when a fuel damaging event occurs.

When the reactor scrams, one scram discharge volume ruptures.

Additional conditions are as follows:

- Drywell radiation monitors are indicating 8 R/hr.
- A Group III isolation has occurred.
- SBGT 'A' failed to start and cannot be started.
- SBGT 'B' has started but a system malfunction has resulted in a reduction in system flow.
- Secondary Containment differential pressure was negative but is now showing positive inches of water.
- The ARM # 2 near the elevator on the 252 foot elevation is pegged high at > 1000 mRem/hr.
- The scram cannot be reset.
- Reactor building access is restricted.
- A team has been dispatched to determine dose at the site boundary.

Based on the information available, the current emergency plan classification is _____ (1) _____.

An Emergency Depressurization is required before the whole body dose rate at the site boundary exceeds _____ (2) _____ mRem/hr TEDE for at least one hour.

(Note: AP 3125, Emergency Plan Classification and Action Level Scheme is provided as a reference.)

A. Alert
100

B. Alert
1000

C. Site Area Emergency
100

D. Site Area Emergency
1000

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The current classification is a SAE IAW EAL FS1.1, Loss or potential loss of any two barriers. Plausible in that Alert EAL FA1.1, Any loss or any potential loss of the Fuel Clad or RCS has also been exceeded because an unisolable primary system is discharging outside the primary containment resulting in an area radiation above its Max Safe Operating Value. The Max Safe radiation level for ARM # 2 is 1000 mRem/hr.

Additionally an Emergency Depressurization is required before the dose rate exceeds the General Emergency level of 1000 mRem/hr. Plausible if the candidate believes that the ED is required before the dose rate exceeds the Site Area Emergency level of100 mRem/hr.

B. Incorrect: The current classification is a SAE IAW EAL FS1.1, Loss or potential loss of any two barriers.

C. Incorrect: an Emergency Depressurization is required before the dose rate exceeds the General Emergency level of 1000 mRem/hr. Plausible if the candidate believes that the ED is required before the dose rate exceeds the Site Area Emergency level of 100 mRem/hr.

D. Correct: The current classification is a SAE IAW EAL FS1.1, Loss or potential loss of any two barriers. A potential loss of the RCS is occurring because an unisolable primary system is discharging outside the primary containment resulting in an area radiation above its Max Safe Operating Value. The Max Safe radiation level for ARM # 2 is 1000 mRem/hr. An actual loss of the Primary Containment barrier is occurring for the same reason.

With a positive pressure in the Secondary Containment and an SDV rupture, an offsite release is occurring, resulting in an entry condition for EOP-4, Rad Release Control (indication that an offsite radioactivity release is underway that may exceed the level requiring an Alert declaration per AP 3125).

Rad Release Control requires that an Emergency Depressurization be performed before the Offsite rad release exceeds the General Emergency EAL. This will occur when field survey results indicate that the dose rate will exceed 1000 mRem/hr and is expected to continue for an hour or more.

Technical Reference(s): AP 3125 EALs (Attach if not previously provided)

EOP-4, Rad Release Control

Proposed References to be provided to applicants during examination: AP 3125, Hot and Cold EAL Charts

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # WTS Bank # 2149 Modified for VY and changed part 2 to ask when ED is required. Original question asked the candidate to determine when the event needed to be upgraded.

New

Question History: Last NRC Exam: Pilgrim 2002

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295020	2.2.37
	Importance Rating		4.6

Equipment Control: Ability to determine operability and / or availability of safety related equipment.

Proposed Question: SRO Question # 85

The plant is operating at rated conditions when an I&C surveillance results in an inadvertent HPCI high steam flow isolation.

- HPCI responds as designed to the isolation signal.
- I&C removes the inadvertent signal and both Isolation Signal Isolation Reset Pushbuttons are depressed.

Under these conditions, the SRO should declare HPCI…

A. INOPERABLE because the HPCI-15 and HPCI-16 valves are closed.

B. INOPERABLE because the HPCI suction re-aligned to the torus.

C. OPERABLE because an initiation signal will automatically re-align HPCI for injection.

D. OPERABLE because operator action can be credited for verifying all valves reposition if an initiation signal occurs.

Proposed Answer: A

Explanation (Optional):

A. Correct: The isolation signal tripped closed the following valves:
   - STEAM ISOLATION HPCI-15
   - STEAM ISOLATION HPCI-16
   - PUMP SUCTION (TORUS) HPCI-57
   - PUMP SUCTION (TORUS) HPCI-58

   Although the isolation has been reset, HPCI must still be considered inoperable because the HPCI-15 valve is closed. IAW OP 2120, Precaution 1, whenever HPCI-15 is closed for other than surveillance testing the HPCI system shall be declared inoperable.

B. Incorrect: HPCI suction did not re-align to the torus as the CST suction did not close. If it had, precaution 24 states that whenever HPCI suction is aligned to the torus it shall be declared inoperable. Plausible if the candidate believes that the isolation signal isolates both suction, and steam supply as it would if aligned to the torus.

C. Incorrect: Although the statement is true, HPCI must still be declared inoperable due to the HPCI-15 valve being closed.

D. Incorrect: Operator action cannot be credited for opening the HPCI-15 valve.

Technical Reference(s): OP-2120 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	223002 A2.01	
	Importance Rating		3.5

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: A.C. electrical distribution failures

Proposed Question: SRO Question # 86

Given the following:

- The plant is in cold shutdown with the 'A' Loop of RHR in SDC.
- A loss of the 120V/240V VAC VITAL Bus then occurs.
- ON 3168, Loss of 120/240 VAC Vital Bus is entered.
- The VITAL Bus is manually re-energized five minutes later.

Which one of the following is correct regarding:

(1) The PCIS system impact?

AND

(2) Required additional procedure actions?

A. (1) No valves close but 'half' an isolation signal occurs for Groups 1, 2, 3, 4 and 5.

(2) Enter OP 2115, Primary Containment, and reset the half isolation signal for Groups 2, 3, 4 and 5.

B. (1) Outboard isolation valves for Groups 2, 3, 4 and 5 close.
Group 1 Outboard valves do not close but receive 'half' an isolation signal.

(2) Enter ON 3156, Loss of Shutdown Cooling.

C. (1) No valves close but 'half' an isolation signal occurs for Groups, 2, 3, 4 and 5.
Group I outboard isolation valves close.

(2) Enter OP 2115, Primary Containment, and reset the half isolation signal for Groups 2, 3, 4 and 5.
Enter OP 2113, Main Steam, reset the isolation and re-open the MSIVs.

D. (1) Outboard isolation valves for Groups 2, 3, 4 and 5 close.
Group 1 Outboard valves do not close but receive 'half' an isolation signal.

(2) Enter ON 3156, Loss of Shutdown Cooling.
Enter OP 2113, Main Steam, close the Outboard MSIVs to allow reset, and reset the 'half' isolation using the Group I Isolation Reset Switch.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Outboard isolation valves close for Groups 2, 3, 4 and 5. Plausible in that a 'half' isolation signal does occur for the MSIVs (Group 1).

B. Correct: Outboard isolation valves close for Groups 2, 3, 4 and 5. Group 1 valves receive a 'half' isolation but do not close. With the RHR-17 valve, SDC Suction, valve closing, the loss of SDC procedure must be entered. Additionally I&C assistance is required to reset the 'half' MSIV isolation as a PCIS relay must be 'fingered' to allow reset. This process is addressed in ON 3168.

C. Incorrect: Outboard isolation valves close for Groups 2, 3, 4 and 5. Group 1 valves do not close but receive 'half' an isolation signal. Additionally it would not be necessary to enter OP 2113, Main Steam and reopen the MSIVs.

D. Incorrect: It is not necessary to enter OP 2113, close the MSIVs in order to reset the isolation. Plausible in that if a full isolation were to occur, this would be necessary.

Technical Reference(s): OP-3168 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	261000 A2.10	
	Importance Rating		3.2

Ability to (a) predict the impacts of the following on the STANDBY GAS TREATMENT SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor water level: Plant-Specific

Proposed Question: SRO Question # 87

Given the following:

- The plant is commencing a shutdown due to an electrical fault on MCC-8B.
- The rest of the electrical distribution system is in its NORMAL lineup.
- A loss of normal power occurs and ALL systems operate as designed.

  Post scram conditions are as follows:

- Water level lowered to 106 inches and is now being controlled in the prescribed post scram level band.
- Drywell temperature is 152°F and rising slowly.
- All control rods fully inserted.

For these given conditions, determine the following:

(1) How is the status of the Standby Gas Treatment (SBGT) system verified?

(2) What is the current status of the SBGT system?

The SBGT system status is checked by verifying the applicable automatic actions of Table ‘A’, "Initiations/Isolations", in accordance with _(1)_

_(2)_ train(s) of the Standby Gas Treatment System will be running.

A. (1) EOP-1, "RPV Control"
(2) BOTH the ‘A' AND ‘B’

B. (1) EOP-1, "RPV Control"
(2) ONLY the ‘A’

C. (1) EOP-3, "Primary Containment Control"

(2) BOTH the ‘A’ AND ‘B’

D. (1) EOP-3, "Primary Containment Control"
(2) ONLY the ‘B’

Proposed Answer: A

Explanation (Optional):

A. CORRECT. :Both trains auto start on low level and are verified IAW Table ‘A’ in EOP-1 override.

B. INCORRECT: power is available to both trains.

C. INCORRECT: Table ‘A’ verification is in EOP-1 only

D. INCORRECT: Table ‘A’ verification is in EOP-1 only; power is available to both trains.

Technical Reference(s): EOP-1 table A (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-610, objective 2.4.43 (As available)

Question Source: Bank # WTSI 14563
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2010 Vermont Yankee

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

10 CFR Part 55 Content: 55.41
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	205000 2.1.27	
	Importance Rating		4.0

Conduct of Operations: Knowledge of system purpose and / or function (Shutdown Cooling)

Proposed Question: SRO Question # 88

An RPV cooldown using shutdown cooling (SDC) is in progress.

Current plant conditions are as follows:

- RPV pressure is 90 PSIG.
- RPV temperature is 330°F.
- 'A' loop of RHR is in SDC mode of operation.
- Current cooldown rate is 80°F/hr.

Engineering then informs the control room that the supplier reported that Shutdown Cooling RPV high reactor pressure switches (PS-2-128 A & B) should be considered inoperable due to a common manufacturing defect.
I&C reports the pressure switch operability will be restored in 36 hours.

Which one of the following is required for the above conditions?

(Note: Reference provided is TS Table 3.2.2 and associated notes.)

A. SDC must be isolated within one hour.

B. If RPV temperature has not been reduced to < 212°F within two hours, SDC must be isolated.

C. RPV temperature must be reduced to < 212°F within the next 24 hours.
No other action is required.

D. SDC can remain is service for 24 hours regardless of RPV temperature.
Both channels must then be tripped.

Proposed Answer: A

Explanation (Optional):

A. Correct: Up to two hours are available before SDC must be isolated or achieve Cold S/D conditions except that it has been reported that switch restoration will be 36 hours negating two hours as an option. Plausible in that Table 3.2.2, Note 1 requires that if isolation capability is not restored within 1 hour then note 2d applies which says that SDC must be isolated. However up to 1 hour is allowed in Note 2d to isolate SDC following the elapse of the 1 hour to restore isolation capability for a total of 2 hours.

B. Incorrect: With both pressure switches inoperable, isolation capability has been lost. Table 3.2.2, Note 1.b, requires that if isolation capability is not restored within 1 hour then note 2 applies except that it has been reported that switch restoration will be 36 hours negating two hours as an option.. Table 3.2.2, requires that when Note 2 is applied, action 2d is applicable for the SDC high pressure isolation. Note 2d requires that SDC must be isolated within the next hour. Therefore only one hour is available before SDC must be isolated or conditions established that do not require isolation capability. This would occur when RPV temperature is < 212 °F.

C. Incorrect: The loss of SDC isolation capability must be addressed as discussed in B above. Plausible if the candidate only considers that Primary Containment integrity requirements are not met and applies the LCO action of section 3.7. This is further reinforced by Note 2b of Table 3.2.2 which directs that the reactor be placed in cold S/D within 24 hours. However that action is not applicable to a loss of the high pressure isolation capability of SDC.

D. Incorrect: Unless cold S/D conditions can be achieved within 2 hours, SDC must be isolated. Plausible in that Note 1directs that with one or more Trip Functions with one or more required channels inoperable that the channel is to be placed in trip within 24 hours. However Note 1 requires that both part ‘a’ and part ‘b’ be addressed. Part ‘b’ addresses loss of isolation capability which is the current situation. Note 1 then directs the operator to Note 2 if part ‘b’ cannot be met which requires the SDC isolation.

Technical Reference(s): TS Table 3.2.2 and associated notes. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: TS Table 3.2.2 and associated notes. No bases

Learning Objective: (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	206000 2.4.2	
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions (HPCI)

Proposed Question: SRO Question # 89

The plant is at rated conditions when an inadvertent Group I isolation occurs.
Additional plant conditions are as follows:

- The reactor was manually scrammed when the RO observed RPV pressure at 1100 psig and rising.
- All rods inserted.
- Highest RPV pressure during the transient was 1300 psig.
- Drywell pressure is 2.7 psig and up slow.
- RPV pressure is currently cycling between 1020 and 1085.
- Lowest RPV level during the transient was 92 inches.
- Highest RPV level during the transient was 172 inches.
- Torus water temperature is 74°F and rising.

Assuming that all systems responded as designed which one of the following is correct immediately following the initial transient described regarding:

(1) The systems that injected into the RPV?

(2) procedures that are now required?

A. (1) Feed System only.
(2) Enter EOP-1 only.

B. (1) Feed System and HPCI.
(2) Enter EOP-1 and EOP-3.

C. (1) Feed System only.
(2) Enter EOP-1 and EOP-3.

D. (1) Feed System and HPCI.
(2) Enter EOP-1 only.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: HPCI is also injecting on via a high drywell pressure initiation system. Also EOP-1 directs that if an SRV is cycling (as indicating by RPV pressure cycling around the lowest SRV setpoint) then SRVs should be opened to lower pressure to < 1055 psig. Additionally HPCI cannot be used for pressure control until drywell pressure lowers.

It would not be necessary to place Recirc pumps at minimum speed as they would already be tripped.

B. Correct: With pressure rising to 1300 psig, both safety valves lifted. Since they relieve to the drywell, pressure will rise to > 2.5 psig causing a HPCI initiation. Additionally, both Recirc pumps will trip via the RPT trip at 1150 psig. Due to the high drywell pressure, entry conditions to both EOP-1 and EOP-3 will be received. EOP-1 directs that if an SRV is cycling (as indicating by RPV pressure cycling around the lowest SRV setpoint) then SRVs should be opened to lower pressure to < 1055 psig.

Since the reactor is on natural circulation, a Recirc pump should be started IAW OP 2110 to prevent thermal stratification.

C. HPCI is also injecting on via a high drywell pressure initiation system. It would not be necessary to place Recirc pumps at minimum speed as they would already be tripped.

D. Incorrect: HPCI cannot be used for pressure control until drywell pressure lowers.

Technical Reference(s): EOP-1 and EOP-3 entry conditions.
LOT-00-239, page 29, regarding Safety Valves
LOT-00-202, page 55 regarding high RPV pressure trip of Recirc pumps.

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	400000 A2.04	
	Importance Rating		3.0

Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Radiation monitoring system alarm

Proposed Question: SRO Question # 90

The plant is operating at 100% power with all conditions normal.

Then, alarm RBCCW EFFLUENT RAD HI (3-E-6) annunciates.

Assuming the alarm is valid, which set of the following responses is correct regarding:

(1) What is another indication that is consistent with this alarm?

AND

(2) What is the appropriate action to direct?

A. (1) RBCCW surge tank high level.

(2) Enter OPON-3147, Loss of RBCCW, align alternate cooling to the CRD and RHR pumps and isolate the RBCCW supply while monitoring surge tank level.

B. (1) Lowering RBCCW system temperature.

(2) Enter OPON-3153, Excessive Radiation Levels, and isolate RCU while monitoring RBCCW system temperature.

C. (1) RBCCW surge tank high level.

(2) Enter OPON-3153, Excessive Radiation Levels, and isolate RCU while monitoring surge tank level.

D. (1) Lowering RBCCW system temperature.

(2) Enter OPON-3153, Excessive Radiation Levels, and shift RBCCW heat exchangers while monitoring RBCCW system temperature.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The loss of RBCCW procedure does not address this condition. Additionally leakage from the CRD and RHR pump coolers would not cause Rad Levels to rise in the RBCCW system during normal operation. Plausible in that the action to shift cooling water supply to the RHR and CRD pumps is a step in this procedure.

B. Incorrect: A lowering RBCCW temperature might occur if service water was leaking into the RBCCW system via the heat exchanger. However it would not cause rad levels to rise in the RBCCW system. Whereas a RCU leak from the NRHX would tend to cause temperatures to rise.

C. Correct: For Rad levels to rise in the RBCCW water from the primary system would have to be leaking into the RBCCW system. This would cause surge tank level to rise. The high rad alarm is a symptom associated with OPON-3153, Excessive Radiation Levels. Step 3.10 of this procedure directs that if the RBCCW radiation monitor indicates a high radiation level, then isolate the RCU system and check surge tank level indication to determine if the leak has been isolated.

D. Incorrect: A lowering RBCCW temperature is not consistent with a rising Rad level in the RBCCW system but might be consistent with a service water leak into the system via the RBCCW Heat Exchanger. Also plausible in that OPON-3153, includes direction to shift RBCCW Heat Exchangers but this is only done if the Service Water Effluent high rad alarm has also annunciated.

Technical Reference(s): OPON-3153-01, Excessive Radiation Levels. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam: Not Used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	201002	A2.01
	Importance Rating		2.8

Ability to (a) predict the impacts of the following on the REACTOR MANUAL CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Rod movement sequence timer malfunctions

Proposed Question: SRO Question # 91

Given the following:

- The reactor is operating at 50% CTP.
- A control rod is selected and notched out one notch from position 32 to 34.
- The rod sequence control timer fails, the withdraw bus remains energized and the control rod continues out.

Based on the above conditions which ONE of the following describes the effect on the system and the procedural guidance that should be used?

The 2 second cycle auxiliary timer will provide a control rod (1)_____.
You should (2)_____.

A. (1) select block, and you should confirm the control rod deselected

(2) use ARS 5-D-6, Rod Timer Malf

B. (1) withdraw block, and you should confirm the control rod stopped moving

(2) use ARS 5-D-6, Rod Timer Malf

C. (1) withdraw block, and you should confirm the control rod deselected

(2) enter OT 3166, Mispositioned Rod

D. (1) select block, and you should confirm the control rod stops moving at position 48

(2) enter OT 3166, Mispositioned Rod

Proposed Answer: A

Explanation (Optional):

A. Correct Response - the control rod is deselected and will settle on the next notch.

B. Incorrect - A select block is applied, not a withdraw block.

C. Incorrect - The control rod should not drift but stop on the next notch. A drift alarm may occur.

D. Incorrect - The control rod is deselected and will settle on the next notch.

Technical Reference(s): ARS 21003, 5-D-6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-02-201, K2.01 (As available)

Question Source: Bank # VY LOR Bank

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	239001	2.1.23
	Importance Rating		4.4

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO Question # 92

During a plant startup an AO is completing a Main Steam valve lineup and discovers a steam leak on MS-6, steam seal regulator inlet.
He has informed the control room that valves in the area to be verified are inaccessible.
For a component to be deemed 'inacessible' due to location IAW AP 0155, CURRENT SYSTEM VALVE AND BREAKER LINEUP AND IDENTIFICATION, which ONE of the following is correct and what are the SRO actions?

The SRO will determine the validity of the inaccessibility which may include _____(1)_____.
Then the SRO will then_____(2)_____.

A. (1) high radiation area such that an operator would receive a dose of 15 mREM or more to verify the component.
(2) transfer component status from the previous lineup in the current system lineup book for those components that are inaccessible.

B. (1) high temperature in the area such that a safety hazard exists.
(2) transfer component status from the previous lineup in the current system lineup book for those components that are inaccessible.

C. (1) high radiation area such that an operator would receive a dose of 15 mREM or more to verify the component.
(2) transfer component status from the previous lineup to the Lineup Deviation Book for those components that are inaccessible.

D. (1) high temperature in the area such that a safety hazard exists.
(2) transfer component status from the previous lineup to the Lineup Deviation Book for those components that are inaccessible.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The Radiation limit is greater than 20 mr.

B. Correct Response- AP-0155, section 4.2.12 and definitions, this is the correct response.

C. Incorrect -The Radiation limit is greater than 20 mr. The lineup deviation book entry would not be warranted.

D. Incorrect –The lineup deviation book entry would not be warranted.

Technical Reference(s): AP-0155, CURRENT SYSTEM VALVE AND BREAKER LINEUP AND IDENTIFICATION (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-01-400 CRO 9 (As available)

Question Source: Bank # VY LOR

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	219000	A2.06
	Importance Rating		2.9

Ability to (a) predict the impacts of the following on the RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: D.C. electrical failures

Proposed Question: SRO Question # 93

Bold items added during examination administration.

Given the following:

- The plant is operating at 98% RTP.
- The 'A' Loop of RHR is in Torus Cooling with the 'A' RHR Pump operating to support the HPCI Pump Operability and Flow Rate Test which is in progress.
- HPCI has been operating for 15 minutes.

With the above conditions present, the plant sustains a loss of DC-2.

Which ONE of the following describes the plant response and procedure to enter?

A. The 'A' RHR Pump trips due to the loss of the valve position indication for the suction valve.
Enter ON 3160**, Loss of DC-2 and DC-3**.

B. The 'A' RHR Pump continues to run with the pump status lights on CRP 9-3 not lit.
Enter ON 3160**, Loss of DC-2 and DC-3**.

C. The 'A' RHR Pump trips due to the loss of the valve position indication for the suction valve.
Enter OPON-3159-01**, Loss of DC-1**.

D. The 'A' RHR Pump continues to run with the pump status lights on CRP 9-3 not lit.
Enter OPON-3159-01**, Loss of DC-1**.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The ‘A’ RHR Pump will not be affected. This would ONLY be the case if the plant was in Shutdown Cooling.

B. Correct: A loss of DC-2 affects the ‘A’ RHR pump in that control power is lost so there is nothing to trip the breaker. ON 3160 addresses the loss of DC-2

C. Incorrect: The ‘A’ RHR Pump will not be affected. This would ONLY be the case if the plant was in Shutdown Cooling. OPON-3159-01 is for a loss of DC-1.

D. Incorrect. A loss of DC-2 affects the ‘A’ RHR pump in that control power is lost so there is nothing to trip the breaker. OPON-3159-01 is for a loss of DC-1.

Technical Reference(s): OP 2145 Appendix C; ON 3160 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: NO

Learning Objective: LOT-00-263 k3.03 (As available)

Question Source: Bank # VY LOR BANK

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		1
	K/A #	G1	2.1.6
	Importance Rating		4.8

Conduct of Operations: Ability to manage the control room crew during plant transients.
Proposed Question: SRO Question # 94

The plant is at rated conditions when a small steam leak in the drywell causes a reactor scram.

When HPCI starts a HPCI steam leak occurs.
HPCI cannot be isolated.

Plant conditions are currently:

- RPV water level is 100 inches, rising slowly.
- Drywell pressure is 3.2 psig, rising slowly.
- RPV pressure is 912 psig with one turbine bypass valve open controlling RPV pressure.
- Reactor Building HVAC has isolated and SBGT is running.

Secondary Containment parameters are as follows:

- Torus SW 213' temperature, Channel 6, is 252°F and rising slowly.
- Torus SE 213' temperature, Channel 8, is 255°F and rising slowly.
- RB SW 252' temperature, Channel 12, is 155°F and rising slowly.
- All other parameters are below their max normal operating values.
- There are no other abnormal indications within the secondary containment or its ventilation system.

Which ONE of the following actions is correct?

(NOTE: EOP-4, Table O is provided for your use.)

A. Enter EOP-5, RPV-ED and open all SRVs.
Direct the crew to defeat isolation interlocks and restore reactor building HVAC.

B. Enter EOP-5, RPV-ED and open all SRVs.
Do NOT direct operators to defeat isolation interlocks to restore reactor building HVAC.

C. Anticipate RPV-ED IAW EOP-1, RPV Control, and open all SRVs.
Do NOT direct operators to defeat isolation interlocks to restore reactor building HVAC.

D. Anticipate RPV-ED IAW EOP-1, RPV Control, and rapidly depressurize the RPV using turbine bypass valves, disregarding cooldown rate.
Direct the crew to defeat isolation interlocks and restore reactor building HVAC.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: RPV –ED is only performed when the same parameter is above its Max Safe Value in 2 or more areas. Plausible in that 2 area temperatures are above max safe but they are in the same area (Torus area). Additionally a second area is also above its max safe value for water level (but a different parameter).

B. Incorrect: RPV-ED is not yet required. Additionally reactor building isolation interlocks are to be defeated and reactor building HVAC restored.

C. Incorrect: The correct action is to anticipate the RPV-ED. However it is not correct to open all SRVs to do so. That action would be counterproductive to the goal of minimizing the heat addition to the containment. Reactor building isolation interlocks are also required to be defeated and reactor building HVAC restored because HVAC is isolated and reactor building vent exhaust is below 14 mr/hr per the current plant conditions of the stem.

D. Correct: EOP-01 directs that if RPV-ED is anticipated then the RPV is to be rapidly depressurized via the turbine bypass valves regardless of the cooldown rate. RPV-ED will be required when RB SW 252' temperature rises an additional 5 °F. Reactor building isolation interlocks are to be defeated and reactor building HVAC restored because HVAC is isolated and reactor building vent exhaust is below 14 mr/hr per the current plant conditions of the stem.

Technical Reference(s): EOP-1, 3rd override of step RC/OR-4
EOP-4, 3rd override of step SC/OR-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-4, Table O

Learning Objective: LOT-00-611 SRO EO 3 (As available)

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

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		2
	K/A #	G2	2.2.23
	Importance Rating		4.6

Equipment Control: Ability to track Technical Specification limiting conditions for operations.

Proposed Question: SRO Question # 95

Bold items added during the exam administration

Given the following:

- The is plant operating at 95% power.
- HPCI Quarterly Valve Stroke Surveillance, OP 4120, is in progress.
- During the performance of this surveillance, HPCI-15 failed to stroke closed in the required time.
- A second attempt to stroke HPCI-15 is also unsuccessful.
- All other plant equipment is operable.

Which one of the following identifies the Technical Specification requirements related to this condition?

A. Tech Specs allow continued operations for 14 days as long HPCI-16 is closed**, de-energized,** and its position is verified once per 31 days.

B. Tech Specs do NOT allow continued operations with HPCI-15 inoperable. Commence a plant shutdown and cooldown to have RPV pressure below 150 PSIG within 31 days.

C. Tech Specs allow continued operations provided HPCI-16 remains open**, energized,** and operable.

D. Tech Specs do NOT allow continued operations with HPCI-15 inoperable. Commence a plant shutdown, and log the position of HPCI-15 daily.

Proposed Answer: A

Explanation (Optional):

A. Correct. Valve in line must be closed and verified isolated.

B. Incorrect. Operation is allowed IAW reference

C. Incorrect. HPCI-16 is closed

D. Incorrect. Operation is allowed

Technical Reference(s): EN-OP-108 eSOMS LCO module (Attach if not previously provided)

TS 3.7.D.2, 4.7.D.2, 3.5.E.2

Proposed References to be provided to applicants during examination: N

Learning Objective: needed (As available)

Question Source: Bank # VY 1197

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

editorial mods for plausibility

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.5
	Importance Rating		2.9

Radiation Control: Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO Question # 96

The plant operating at full power with the following conditions:

- AOG dryer skid and adsorber bed bypass valves OG-145 and OG-146 were opened on the previous shift.
- AOG system outlet radiation monitor RAN-OG-3127 has just failed downscale.
- Annunciator 50-M-2, AOG SYSTEM OUT RAD MON TROUBLE, has just alarmed.
- OG-FCV-11, Off Gas to Stack Isolation, is open.
- All the stack monitoring instrumentation is operable.

Which ONE of the following actions is required?

(NOTE: ODCM Section 3.1.2 is provided.)

A. (1) Bypass RAN-OG-3127 to prevent closure of Stack Isolation OG-FCV-11.

(2) Enter a tracking LCO.

B. (1) Immediately close OG-145 and OG-146.

(2) Enter a tracking LCO.

C. (1) Bypass RAN-OG-3127 to prevent closure of Stack Isolation OG-FCV-11.

(2) Operation may continue for 7 days provided the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

D. (1) Immediately close OG-145 and OG-146.

(2) Operation may continue for 7 days provided the stack monitoring systems is operable and off-gas system temperature and pressure are measured continuously.

Proposed Answer: A

Explanation (Optional):

A. Correct – When the AOG dryer skid and adsorber bed bypass valves OG-145 or OG-146 are open and either AOG monitor fails downscale OG-FCV-11 will automatically close unless bypassed within 2 minutes. ON-3152, MSL and Off Gas High Radiation, directs placing the keylock switch for RAN-OG-3127 to TEST to bypass the isolation. IAW ODCM, Section 3.1.2, The minimum number of AOG system outlet radiation monitor channels required to be operable is 1 and all stack monitoring instruments are operable therefore no ODCM actions are required. Additionally NO ODCM actions are required if OG-145 and OG-146 are open less than seven days

B. Incorrect – There is NO requirement to immediately close OG-145 and OG-146.

C. Incorrect –IAW ODCM, Section 3.1.2, The minimum number of AOG system outlet radiation monitor channels required to be operable is 1. Regarding Note 2 of Table 3.1.2, all stack monitoring instruments are operable therefore no ODCM actions are required.

D. Incorrect – There is NO requirement to close OG-145 and OG-146. IAW ODCM, Section 3.1.2, The minimum number of AOG system outlet radiation monitor channels required to be operable is 1. Regarding Note 2 of Table 3.1.2, all stack monitoring instruments are operable therefore no ODCM actions are required.

Technical Reference(s): ARS-21018, 50-M-2
ON-3152, pg 3
ODCM Section 3.1.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: ODCM section 3.1.2

Learning Objective: (As available)

Question Source: Bank # VY

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4 2.4.18	
	Importance Rating		4.0

Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: SRO 97

The plant was operating at full power when an event occurred.
The operators inserted a reactor scram.

The following conditions exist:

- Reactor power is at 9%
- RPV level is at -28 inches and lowering
- 3 SRVs are open

The operators are performing an RPV-ED.
RPV pressure is now at 350 psig and lowering.

Which one of the following describes;

(1) What pressure must the RPV be lowered to before the CRS orders reinjecting to the vessel ?

AND

(2) what the Minimum Steam Cooling Pressure (MSCP) limit precludes?

(Note: EOP-2 MSCP table provided)

A. (1) 215 psig
(2) it precludes any clad temperature from exceeding 1500°F **even if** the core is uncovered.

B. (1) 215 psig
(2) it precludes any clad temperature from exceeding 1500°F **unless** the core is completely uncovered.

C. (1) 160 psig
(2) it precludes any clad temperature from exceeding 2200°F **even if** the core is uncovered.

D. (1) 160 psig
(2) it precludes any clad temperature from exceeding 2200°F **unless** the core is completely uncovered.

Proposed Answer: A

Explanation (Optional):

A: Correct.

B: Incorrect - Per EOP ATWS step ARC/L-13 bases it precludes any clad temperature from exceeding 1500°F even if the core is uncovered

C: Incorrect - Per EOP ATWS – RPV Control, the MSCP with 3 open SRVs is 215. 160 psig is for 4 SRVs

D: Incorrect - it precludes any clad temperature from exceeding 1500°F even if the core is uncovered. 160 psig for 4 SRVs

Technical Reference(s): EOP-2 ATWS RPV control
EOP Study Guide (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-2 MSCP table

Learning Objective: (As available)

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

Question History: 2010 Duane Arnold

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

10 CFR Part 55 Content: 55.41

55.43 1

Conditions and limitations in the facility license

Comments:

Editorial plant specific mods

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		4
	K/A #	G4	2.4.46
	Importance Rating		4.2

Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions.

Proposed Question: SRO Question # 98

A reactor plant startup and heatup is in progress.

Initial plant conditions are:

- All IRMs are on range 8.
- The reactor mode switch is in STARTUP.

Then, SRM 'D' fails resulting in the following indications:

- Annunciator SRM HI/INOP, 5-P-3, alarms.
- Annunciator, Rod Withdraw Block, 5-D-3, remains clear.
- SRM 'D' is pegged high on both the back panel and CRP 9-5 panel indications.

Which ONE of the following should the CRS direct and the reason for that action?

A. Manually insert a Rod Block immediately.
The Rod Block circuit failed to trip following the SRM failure.

B. Manually insert a Rod Block within the next hour.
The Rod Block circuit failed to trip following the SRM failure.

C. Place SRM 'D' in the tripped condition within 12 hours.
There are an insufficient number of operable SRMs.

D. No Tech Spec/TRM actions are currently required.
SRM rod blocks are not required given the initial plant conditions.

(Note: TRM Table 3.2.5 and associated notes provided)

Proposed Answer: D

Explanation (Optional):

A. Incorrect: With all IRMs on Range 8 or above SRM rod withdrawal blocks are not enforced on SRM upscale trips.

B. Incorrect: With all IRMs on Range 9 or above SRM rod withdrawal blocks are not enforced on SRM upscale trips.

C. Incorrect: The SRM function is not required with the IRMs on range 8. Plausible in that TRM Table specifies 3.2.5 specifies that the minimum number of channels for SRM High Rod Block is 4. With only three remaining, a 12 hour LCO would be required if the SRMs if the IRMs were on a lower range.

D. Correct: SRM Hi rod blocks are bypassed with all IRMs are on Range 8. Since the SRMs are not required no Tech Spec/TRM actions are required.

Technical Reference(s): TRM Table 3.2.5 (Attach if not previously provided)

OP 2130 page 3

Proposed References to be provided to applicants during examination: TRM Table 3.2.5 and associated notes

Learning Objective: (As available)

Question Source: Bank # X Modified for VY

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: Pilgrim 2009

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference: Level RO SRO

Tier #		3
Group #		2
K/A #	G2	2.2.19
Importance Rating		3.4

Equipment Control: Knowledge of maintenance work order requirements.

Proposed Question: SRO Question # 99

Given the following:

- The plant is at rated conditions.
- 'A' RHR pump has just tripped during a routine surveillance.
- The cause of the pump trip cannot be immediately determined and corrected.
- All other systems are operable.

Which one of the following is correct regarding the prioritization of the associated work order?

The Shift Manager should characterize the work order as which ONE of the following?

A. Priority 1 and direct immediate start of repair efforts in parallel with the initiation and planning of a work order.

B. Priority 1 and direct repair efforts are conducted around the clock following the planning of the work order.

C. Priority 2 and direct repair efforts are conducted around the clock following the planning of the work.

D. Priority 2 and the work week schedule adjusted to accommodate repair.
If repairs cannot be completed before exceeding 50% of the allowable LCO time, the priority shall be upgraded to Priority 1.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This action is ONLY authorized if the maintenance is characterized as Emergency Maintenance. . IAW EN-WM-100, the Shift Manager can then authorize the immediate start of repair efforts, in parallel with initiation and planning of a Priority 1

Work Request/Work Order. The definition of Emergency Maintenance is:

The correction of a condition or deficiency that:

- ☐ Constitutes an immediate and direct threat to the health and safety of the public.
- ☐ Requires immediate attention to prevent deterioration of plant conditions to a possible unsafe or unstable level, which would then constitute an immediate and direct threat to the health and safety of the public.
- ☐ Poses a significant industrial hazard that must be corrected immediately to prevent or mitigate actual serious injury or death.

The pump failure requires entry into an LCO but does not meet the criteria for Emergency Maintenance

B. Correct: Tech Spec 3.5.A.4 and associated bases requires that LPCI be declared inoperable as all active components are required to operable in order for LPCI to be operable. Per EN-WM-100, Attachment 9.1, a failure or significant degradation with a system that requires entry into a Tech Spec AOT, a Priority 1 work order is required. Per page 7 of EN-WM-100, Priority 1 work orders are to be worked around the clock following the planning of the work order.

C. Incorrect: Per EN-WM-100, Attachment 9.1, a failure or significant degradation with a system that requires entry into a Tech Spec AOT, a Priority 1 work order is required

D. Incorrect: Per EN-WM-100, Attachment 9.1, a failure or significant degradation with a system that requires entry into a Tech Spec AOT, a Priority 1 work order is required

Technical Reference(s): EN-WM-100, Work Request (WR) Generation, Attachment 9.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: N

Learning Objective: LOT-00-400 EO 3 (As available)

Question Source: Bank # WTSI 12880

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2011 Pilgrim

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	K/A #	G3	2.3.12
	Importance Rating		3.7

Radiation Control: Knowledge of Radiological Safety Principles pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: SRO Question # 100

The plant is operating at 100% power and moving irradiated fuel in the fuel pool when the following events occur:

- A fuel bundle is dropped and damaged.
- The RM-17-453A/B Rad Monitors (Reactor Building) each indicate 17 mr/hr.

  The following annunciators are in alarm:

  - 3-E-3, RX BLDG RAD HI
  - 5-H-1 and 5-J-1, RX BLDG/REFUEL FLR CH A(B) RAD HI
  - 3-E-4, REFUEL FLOOR RAD HI

Which ONE of the following is required?

A. Enter EOP-4, Secondary Containment and Radioactivity Release Control, ONLY then determine Refuel Floor and Reactor Building radiation levels and direct Chemistry to obtain stack release samples.

B. Enter EOP-4, Secondary Containment and Radioactivity Release Control, and ON-3152, MSL and Off Gas High Radiation, then direct an evacuation of personnel and verify OG FCV 11, Off Gas to Stack Isolation.

C. Enter ON-3152, MSL and Off Gas High Radiation, and ON-3153, Excessive Radiation Levels, then re-start Reactor Building HVAC using OE-3107, Appendix AA and direct Chemistry to obtain stack release samples.

D. Enter ON-3153, Excessive Radiation Levels, and EOP-4, Secondary Containment and Radioactivity Release Control, then direct an evacuation of personnel and request Radiation Protection to obtain area dose rates and air samples.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - There are entry requirements for ON-3153 and an evacuation is required.

B. Incorrect - There are no entry conditions for ON 3152, and no requirement to verify OG FCV 11 isolation for these conditions.

C. Incorrect - There are no entry conditions for ON 3152 and no requirements to re-start RB HVAC at this time or obtain stack samples.

D. Correct - There are entry conditions for ON-3153, Excessive Radiation Levels and EOP-4 Secondary Containment and Radioactivity Release Control, and they direct an evacuation of personnel and request Radiation Protection to obtain area dose rates and air samples.

Technical Reference(s): EOP-4
OPON-3153-01, pg 2
ARP-21001, 3-E-3 and 3-E-4
ARP-21003, 5-H-1 and 5-J-1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: LOT-00-611, SRO-3 (As available)

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41
55.43 4

Radiation hazards that may arise during normal and abnormal situations, including maintenance activities and various contamination conditions.

Comments: