Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Topic and K/A #	209001	K1.14
	Importance Rating	3.7	

Knowledge of the physical connections and/or cause- effect relationships between LOW PRESSURE CORE SPRAY SYSTEM and the following: Reactor vessel

Proposed Question: RO Question # 1

The plant is at rated conditions when the following Core Spray alarm annunciates:

C903L-C7, Injection Header Break Detection

Which of the following Core Spray injection header break locations would result in this alarm?

Location 1: Inside the vessel shroud
Location 2: Between the reactor vessel wall and the shroud
Location 3: Inside the drywell but outside the reactor vessel

Locations ...

A. 1 only

B. 3 only

C. 1 or 2

D 2 or 3

Proposed Answer: D

A. Incorrect: If a CS line breaks inside the shroud, the dPIS low pressure side will detect reactor pressure inside the shroud as usual. Since the alarm is on a high DP, the CS sparger fracture inside the shroud will not cause an alarm

B. Incorrect: A break between the reactor vessel wall and the shroud will also cause the alarm. See “D” below.

C. Incorrect: A break inside the shroud will not cause the alarm. Additionally a break inside the drywell will cause the alarm.

D. Correct: Downstream of the manual isolation valve (6A/B), each core spray system has an instrument line that is connected to the low pressure side of a differential pressure switch, located on instrument rack 2207. These switches (dPIS-1459A/B) provide an alarm on a high DP condition when a core spray line breaks downstream of the injection line check valve. The high pressure side of the dPIS is connected to the standby liquid control injection "outer" pipe, which detects the pressure in the bypass region above the core plate.

If a CS line breaks inside the shroud, the dPIS low pressure side will detect reactor pressure inside the shroud as usual. The CS sparger fracture inside the shroud will not cause an alarm and the CS system can perform a flooding function but its spray will not provide full core spray coverage. The redundant system will provide 100 percent core spray coverage.

If the CS line breaks outside the core shroud, but inside the reactor vessel, the pressure on the low side is now the pressure outside the core shroud. There is an additional pressure drop (about 7.5 psi) across the steam separators and dryers. If we assume that normal operating sensed dP is -3.0 psid and the low side pressure is decreased by 7.5 psi, we obtain +4.5 psid. The alarm setpoint is -2.5 to 0.5 psid, and will therefore cause an alarm in the control room. When a break outside the reactor vessel but inside the drywell occurs, the pressure on the low side becomes drywell pressure (about 1.1 psig). Any reactor pressurization will cause (high-low) to exceed the alarm setpoint. Therefore, a core spray line break in the drywell will also cause the control room alarm, along with indications of a break in the drywell

Technical Reference(s)	Core Spray Reference Text, page 18 and figure 6	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-09-02, EO-12		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:		N/A

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	
	Topic and 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 # 2

The plant is at 100% power when a loss of 125VDC Bus D-17 occurs.

While investigating the DC buss loss, drywell pressure rises to 2.5 psig.

If NO Operator action is taken, what will be the status of 4KV buses A5 and A6 one minute later?

A. A-5 energized by EDG “A”
A-6 de-energized

B. A-5 energized by the Startup Transformer
A-6 energized by EDG “B”

C. A-5 energized by the Startup Transformer
A-6 energized by the Shutdown Transformer

D A-5 energized by the Startup Transformer
A-6 de-energized

Proposed Answer: D

A. Incorrect: A-5 will be energized by the Startup Transformer. Plausible in that EDG”A” will receive a start signal on high drywell pressure. However the logic associated with the breakers will preferentially choose the Startup Transformer as the A-5 supply.

B. Incorrect: Bus A-6 will be de-energized. Plausible if the candidate does not recognize that EDG “B” starting logic and output breaker control power is also lost.

C. Incorrect: Bus A-6 will be de-energized. Plausible in that the control power for the Shutdown Transformer output, breaker 802, is supplied from DC Bus D-6. However the Shutdown Transformer supply to A-6, breaker 601, is supplied from D-5.

D. Correct: Bus D-17 supplies 125 VDC bus D-5. D-5 supplies EDG “B” start logic and control power to bus A-6 supply breakers. When the reactor scrams on high drywell pressure and the turbine trips, bus A6 will de-energize. Bus A5 which still has control power, will fast transfer to the Startup Transformer.

Technical Reference(s)	Emergency AC Distribution Ref Text, Figure 1, PNPS 5.3.12, Attachment 1, Caution 3. PNPS 2.2.14, Attachment 5	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective: O-RO-02-01-02 EO 8 (As available)

Question Source:	Bank		
	Modified Bank	X	See Comments
	New		

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	
	55.43	7

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

Comments: This modified question is based on a bank question used on the 2014 PNPS NRC exam as RO question # 50. Added to the stem that drywell pressure rose to 2.5 psig. Modified two distractors to address the EDGs. Utilizing the EDG status as possible distractors is plausible due to the EDG start signal generated by the high drywell pressure condition.

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Topic and K/A #	263000	K2.01
	Importance Rating	3.1	

Knowledge of electrical power supplies to the following: Major D.C. loads

Proposed Question: RO Question # 3

The plant is at full power with both the HPCI and RCIC systems in their normal standby lineups.

Then a loss of 250V DC Bus D10 occurs.

With the failure of this bus, which one of the following is correct regarding the ability to operate HPCI and RCIC from the control room?

A. HPCI System cannot be operated in any mode; RCIC is operable in all modes.

B. HPCI System can still be operated in the injection mode; RCIC System is operable in all modes.

C. HPCI System can still be operated in the injection mode; RCIC System can only be operated in the injection mode.

D HPCI System cannot be operated in any mode; RCIC can only be operated in the injection mode.

Proposed Answer: D

A. Incorrect: RCIC cannot be operated in the pressure control mode as HPCI valve, MO-2301-15, HPCI/RCIC TEST RETURN, is without power.

B. Incorrect: HPCI cannot be operated in the injection mode because the valves that need to be opened for injection are without power. Additionally the Aux Oil pump is without power. Also RCIC cannot be operated in the pressure control mode as discussed in "D" below.

C. Incorrect: HPCI cannot be operated in the injection mode because the valves that need to be opened for injection are without power. Additionally the Aux Oil pump is without power.

D. Correct: A loss of D10 will also result in a loss of 250V DC bus, D-9. Multiple HPCI valves will lose 250V DC motive power with the valves in their normal standby and closed position. The HPCI Aux oil pump will also be without power. This will prevent HPCI from being operated in either the Pressure Control Mode or the injection mode.

RCIC can still be operated in the injection mode as all components needed for injection are powered from the "A" 125V DC Battery. However to place RCIC in pressure control mode, HPCI valve, MO-2301-15, HPCI/RCIC TEST RETURN, must be opened. This valve is closed in the HPCI Standby lineup and is now without power. Therefore RCIC cannot be placed in pressure control mode from the control.

Technical Reference(s)	5.3.30 LOSS OF 250V DC POWER BUS D10, Attachment 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-09-03, HPCI, EO-15m		(As available)
Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41		
	55.43	7	

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	
	Topic and K/A #	215005	K2.02
	Importance Rating	2.6	

Knowledge of electrical power supplies to the following: APRM channels

Proposed Question: RO Question # 4

Which one of the following is correct regarding the power supply to the following APRM components?

- APRM instrument drawers
- APRM recorders on C905

	Instrument drawers	**Recorders**
A.	Y-1 Instrument Bus	Y-2 Vital Bus
B.	RPS Power	Y-2 Vital Bus
C.	Y-2 Vital Bus	RPS Power
D	RPS Power	Y-1 Instrument Bus

Proposed Answer: B

A. Incorrect: RPS buses supply the APRM drawers.

B. Correct: 120 VAC from the RPS buses supply the APRM drawers. Y-2 powers the recorders on C905.

C. Incorrect: RPS buses supply the APRM drawers. Y-2 powers the recorders on C905.

D. Incorrect: Y-2 powers the recorders on C905.

Technical Reference(s)	O-RO-02-07-04, slide 49	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-07-04, APRM, EO-6	(As available)
Question Source:	Bank	

	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	206000	K3.02
	Importance Rating	3.8	

Knowledge of the effect that a loss or malfunction of the HIGH PRESSURE COOLANT INJECTION SYSTEM will have on following: Reactor pressure control: BWR-2,3,4

Proposed Question: RO Question # 5

Following a loss of offsite power and reactor scram, HPCI is placed in pressure control mode in order to conduct a plant cooldown.

During the cooldown, the following alarm is received: HPCI INVERTER FAILURE, C903C-A4. The alarm does not reset.

Which one of the following is correct regarding:

(1) The response of RPV pressure to this malfunction AND

(2) An action that should be taken to re-establish control of the cooldown?

A. (1) Pressure will increase.
(2) Place HPCI controller in MANUAL and re-establish the cooldown by manually increasing HPCI flow.

B. (1) Pressure will increase.
(2) Secure HPCI and shift pressure control to the SRVs. Placing HPCI in manual will NOT be effective.

C. (1) Pressure will decrease at a faster rate.
(2) Place HPCI controller in MANUAL and re-establish control of the cooldown by manually decreasing HPCI flow.

D (1) Pressure will decrease at a faster rate.
(2) Secure HPCI and shift pressure control to the SRVs. Placing HPCI in manual will NOT be effective.

Proposed Answer: B

A. Incorrect: The inverter failure will result in a loss of power to the HPCI flow controlling causing HPCI flow to lower to minimum. Because the controller has lost power, placing the controller in manual will not be effective.

B. Correct: The inverter failure will result in a loss of power to the HPCI flow controlling causing HPCI flow to lower to minimum. This will result in less steam demand and pressure to increase. Placing HPCI in manual will not be effective due to the loss of power to the controller and another means of pressure control must be established.

C. Incorrect: Pressure will begin to increase due to the reduction in steam demand. Plausible if the operator believes flow will fail to maximum when the inverter fails. Since the controller has lost power, placing the controller in manual will not be effective.

D. Incorrect: Pressure will begin to increase due to the reduction in steam demand.

Technical Reference(s)	PNPS 2.2.21, section 4.3, page 11		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-09-03, HPCI, 15I		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	261000	K3.06
	Importance Rating	3.0	

Knowledge of the effect that a loss or malfunction of the STANDBY GAS TREATMENT SYSTEM will have on following: Primary containment oxygen content: Mark-I&II

Proposed Question: RO Question # 6

The plant is starting up. Plant conditions are as follows:

- Drywell inerting is in progress
- Standby Gas Treatment (SBGT) train "A" is in service
- Oxygen concentration is at 10% and lowering slowly

Then, the heater for SBGT "A" fails (current flow through the heater falls to zero)

If NO operator action is taken, which one of the following is correct regarding the Oxygen concentration?

Oxygen concentration will ...

A. lower UNTIL Drywell pressure rises to 1.76 psig and then remain constant.

B. lower and continue to lower because SBGT "B" will auto start maintaining flow.

C. lower and continue to lower because the heater loss will have no impact following the manual SBGT start.

D stop lowering immediately and remain constant because SBGT "A" fan will trip and SBGT train "B" will not start.

Proposed Answer: A

A. Correct: The loss of the heater will cause SBGT "A" fan to trip. SBGT fan "B" will not auto start as the low flow start of the standby train is only functional following an auto initiation of SBGT. N2 will continue to flow to the drywell causing O2 concentration to lower. When drywell pressure rises to 1.76 psig, PCV 5030B will close terminating the N2 flow to the drywell.

B. Incorrect: SBGT fan "B" will not auto start as the low flow start of the standby train is only functional following an auto initiation of SBGT.

C. Incorrect: The loss of the heater will cause the fan to trip regardless of the start mechanism (manual or auto).

D. Incorrect: O2 concentration will continue to lower as N2 is still flowing to the drywell. This will continue until the N2 supply is terminated by PCV 5030B closes on high drywell pressure.

Technical Reference(s)	PNPS 2.2.70, Attachment 8 for Inerting lineup PCAC Ref Text, page 12 for discussion of PCV-5030B SBGT Ref Text, page 15 for discussion of SBGT auto start	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-08-02, PCAC, EO-23e		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	215004	K4.04
	Importance Rating	2.8	

Knowledge of SOURCE RANGE MONITOR (SRM) SYSTEM design feature(s) and/or interlocks which provide for the following: Changing detector position

Proposed Question: RO Question # 7

The plant is at rated conditions when a sustained loss of 120VAC Instrument Bus Y-1 occurs. A manual scram is then inserted.

Which one of the following is correct regarding the ability to insert the Source Range Monitor detectors? Do not consider any future actions taken inside the drywell.

A. The detectors cannot be inserted from ANY location because the drive motors are de-energized.

B. The detectors cannot be inserted from ANY location because the drive relays are de-energized.

C. The detectors can ONLY be inserted by actuating the drive relays at a local panel in the Reactor Building.

D The detectors can ONLY be inserted by actuating the drive relays at a local panel in the Cable Spreading Room.

Proposed Answer: C

A. Incorrect: The drive motors are powered from 17L. However the control circuit and relays that actuate the drive motors are powered from Y-1.

B. Incorrect: The drive relays can be actuated from a local panel in the reactor building.

C. Correct: The loss of Y-1 disables SRM detector drive relay control. However capability is provided to manually actually the drive relays from local panel C2214 located in the Reactor Building 23 foot elevation. This action is directed by PNPS 5.3.7.

D. Incorrect: The relay cabinet is not in the Cable Spreading Room but in the Reactor Building.

Technical Reference(s)	PNPS 5.3.7, page 8 item [5], Attachment 2, Item [5]	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-07-01, SRM, EO-20B	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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	
	Topic and 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 # 8

Following a scram RCIC is operating in pressure control mode. The RCIC flow controller is in AUTO and set to control flow at 350 gpm.

If the RCIC System Injection Mode pushbutton is depressed, which one of the following correctly describes CHANGES in the system alignment and flow?

A. MO-1301-49, RCIC PUMP DISCHARGE INJECTION VALVE #2, will open. RCIC flow will increase to 400 GPM.

B. MO-1301-49, RCIC PUMP DISCHARGE INJECTION VALVE #2, will open. MO-1301-53, RCIC FULL FLOW TEST VALVE will close.

C. MO-1301-48, RCIC PUMP DISCHARGE INJECTION VALVE #1, will open. MO-1301-49, RCIC PUMP DISCHARGE INJECTION VALVE #2, will open.

D MO-1301-48, RCIC PUMP DISCHARGE INJECTION VALVE #1, will open. MO-1301-49, RCIC PUMP DISCHARGE INJECTION VALVE #2, will open. MO-1301-53, RCIC FULL FLOW TEST VALVE will close.

Proposed Answer: B

A. Incorrect: The 53 valve will close. RCIC flow will not change because the controller is in AUTO. Plausible in that 400 gpm is rated flow.

B. Correct: The 49 valve will open and the 53 valve will close to divert all pump discharge to the RPV.

C. Incorrect: The 48 valve will not change position. The 48 valve is normally open and must be open for RCIC to be in pressure control mode (return line to CST is downstream of the 48 valve). Additionally, the 53 valve will close to divert all pump discharge to the RPV.

D. Incorrect: The 48 valve will not change position. The 48 valve is normally open and must be open for RCIC to be in pressure control mode.

Technical Reference(s)	RCIC Ref Text Fig 1 (for normal position of RCIC 48 valve and location) PNPS 2.2.22 page 9 and section 4.3, item [3]	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-09-04, EOs-6i & 9	(As available)
Question Source:	Bank Modified Bank New X	(Note changes or attach parent)
Question History:	Last NRC Exam: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 55.43	7

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

Comments:

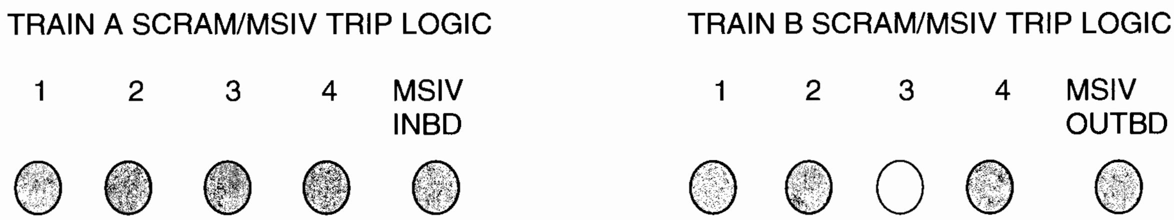
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Topic and K/A #	212000	K5.02
	Importance Rating	3.3	

Knowledge of the operational implications of the following concepts as they apply to REACTOR PROTECTION SYSTEM : Specific logic arrangements

Proposed Question: RO Question # 9

The reactor is operating at 100% power. The following indications are noted on the 905 Panel:

Note: Grey shading indicates the light is illuminated.



The cause is NOT a bad light bulb.

IF APRM ___ (1) ___ (Channel "D" / Channel "E") were to fail upscale, ___ (2) ___ (One-half / One-quarter) of the control rods would scram.

A. (1) Channel "D"
(2) One-half

B. (1) Channel "D"
(2) One-quarter

C. (1) Channel "E"
(2) One-half

D (1) Channel "E"
(2) One-quarter

Proposed Answer: D

A. Incorrect. APRM Channel "D" inputs into RPS "B". The B scram pilot solenoid valves for the GP3 rods are already de-energized. No rods would scram.

B. Incorrect. APRM Channel "D" inputs into RPS "B". The B scram pilot solenoid valves for the GP3 rods are already de-energized. No rods would scram.

C. Incorrect. One-quarter of the rods would insert.

D. Correct: APRM Channel “E” inputs into RPS “A”. ANY A side RPS trip will de-eenergize the A scram pilot solenoid valves for all four Groups of rods. Since the B solenoids for Group 3 rods are already de-energized, the Group 3 rods will scram. Each group comprises approximately 1/4 of the rods

Technical Reference(s)	RPS Ref Text, figures 4 and 12		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-07-07, EO-03j		(As available)
Question Source:	Bank Modified Bank New	WTSI 12611	Editorial Changes
Question History:	Last NRC Exam:	2009	Hope Creek
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		 X
10 CFR Part 55 Content:	55.41 55.43		7

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	
	Topic and 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

The bubbler tube used to measure the level in the Standby Liquid Storage Tank becomes completely clogged.

Regarding the above, complete the following two statements:

(1) The level indication in the control room will indicate …

AND

(2) Local tank level indicator LI-1156 …

A. 1) upscale.
2) CAN be utilized to determine tank level.

B. 1) upscale.
2) CANNOT be utilized and an alternate means of determining tank level must be established.

C. 1) downscale.
2) CAN be utilized to determine tank level.

D 1) downscale.
2) CANNOT be utilized and an alternate means of determining tank level must be established.

Proposed Answer: B

A. Incorrect: Local indication is not available using LI-1156. LI-1156 uses the same bubbler level detector (see "B") and level transmitter as the control room indicator and will also fail upscale.

B. Correct: A bubbler type level detector uses supplied air pressure and a dip tube to determine level of fluid in a tank. An airline is submerged in the tank. The end of the tube is open. Air at low pressure is constantly fed through the tube and bubbles out of the end. The back pressure of the liquid height is read at a pressure transmitter.

With the tube completely plugged, air pressure will rise to the pressure of the air supply. This will be the equivalent of a high back pressure from the level in the tank and level indication will fail high.

Local level indicator LI-1156 is also unavailable for tank level measurement as it uses the same bubbler level detector and level transmitter as the control room indicator. Alternate means must be utilized to determine level.

C. Incorrect: The level indication will fail high. Plausible if the operator does not understand how a bubbler works. Additionally, local indication is not available using LI-1156. LI-1156 uses the same bubbler level detector and level transmitter as the control room indicator.

D. Incorrect - The level indication will fail high.

Technical Reference(s)	SBLC Reference text, Figure 1 P& ID M249		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-06-06, EO-23b		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	262002	K6.01
	Importance Rating	2.7	

Knowledge of the effect that a loss or malfunction of the following will have on the UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) : A.C. electrical power

Proposed Question: RO Question # 11

PNPS is at rated conditions when the following indications are received:

Alarm Y-2 AUTOMATIC TRANSFER (C3RC-A2) annunciates
Alarm Y-1/Y-2 BACKUP POWER SUPPLY TRIP (C3RC-C1) annunciates

Vital AC Bus, Y-2 is …

A. De-energized

B. Energized by B-15

C. Energized by the MG set, powered from B-6

D Energized by the MG set, powered from D-10

Proposed Answer: A

A. Correct – Annunciator C3RC-A2 annunciates when Bus Y2 auto-transfers, annunciator C3RC-C1 annunciates when Breaker 52-1514 is open. ASCO Transfer Switch 83-4 is an automatic dead-bus transfer switch with manual reset. It is used to connect Panel Y2 either to its preferred or normal supply (MG set) or to its emergency or alternate supply (480 VAC power center B15). The ASCO Transfer Switch operates automatically to switch Y2 over into the alternate source of 480V power center Breaker 52-1514. The annunciators indicate that the transfer was attempted but the backup supply breaker has tripped.

B. Incorrect – The annunciator indicate that the transfer was attempted but the backup supply breaker is open.

C. Incorrect – Annunciator C3RC-A2 annunciates when Bus Y2 auto-transfers from the MG Set.

D. Incorrect – Annunciator C3RC-A2 annunciates when Bus Y2 auto-transfers from the MG Set.

Technical Reference(s)	ARP-C3RC, A-2, C-1 2.2.16, Sect 4.1 [2], pg 8	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-01-07, EO-5b		(As available)
Question Source:	Bank	X	PNPS LOR Bank
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	215003	K6.02
	Importance Rating	3.6	

Knowledge of the effect that a loss or malfunction of the following will have on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM : 24/48 volt D.C. power: Plant-Specific

Proposed Question: RO Question # 12

The plant is starting up following a forced outage. Plant conditions are as follows:

- Reactor Mode Switch is in STARTUP, about to be transferred to RUN
- Reactor power is 6% on all APRMs
- All IRMs are on Range 9

Then, 24 VDC Panel D-26 is lost.

Which one of the following is correct regarding the effect of the power loss?

A. IRM Downscale alarm Only

B. IRM Downscale alarm and Rod Block Only

C. RPS "A" Half scram AND Rod Block

D RPS "B" Half Scram AND Rod Block

Proposed Answer: D

A. Incorrect: INOP trips are generated on all "B" side IRMs. This will cause a rod block and half scram. Plausible in that the IRM trips are normally bypassed at high power (3%). However the mode switch must also be in Run for the bypass to occur. With all APRMs reading 6%, the operator may conclude that the INOP trip (and downscale rod block) is bypassed.

B. Incorrect: INOP trips are generated on all "B" side IRMs. This will cause a rod block and half scram. Plausible in that the loss of power will cause the IRMs to fail downscale.

C. Incorrect: A half scram is generated on RPS "B". Plausible if the operator confuses the IRMs that D-26 supplies.

D. Correct: D-26 supplies 24 VDC to IRMs B, D, F, and H. These IRMs input into RPS “B”. When D-26 is lost an INOP trip is generated on all of these IRMs. This in turn will cause a half scram and rod block. The INOP trips are not bypassed until the companion APRMs are > 3% and the mode switch is in Run.

Technical Reference(s)	PNPS 2.2.25, section 4.4	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-07-02, IRM LP, EO-4 and EO-10		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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.

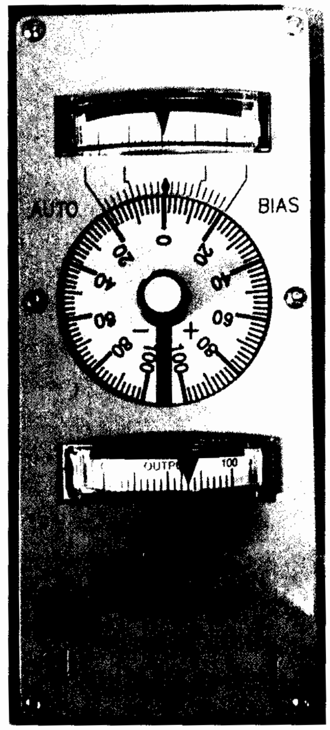
Comments:

Examination Outline Cross-reference:			
	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Topic and K/A #	259002	A1.01
	Importance Rating	3.8	

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR WATER LEVEL CONTROL SYSTEM controls including: Reactor water level

Proposed Question: RO Question # 13

A plant startup is in progress. Additional information is as follows:



- At 10% power the Feedwater Level Control (FWLC) was placed in Master Auto using Feedwater Control Valve "A" IAW PNPS 2.2.82, REACTOR VESSEL WATER LEVEL CONTROL SYSTEM
- FWLC is in Single Element Control

Power is now 30%.

- The current indications on the "A" FLOW Control VALVE Controller, FIC-640-19A, are shown on the picture to the right.
- NO operator actions have been taken regarding the FWLC control system during the power ascension,

Which one of the following would be correct regarding the response of reactor water level if:

The Lever Switch on the controller was moved DIRECTLY from the AUTO position to the MAN position?

Reactor water level would ....

A. Increase

B. Decrease

C. Remain the same

D Cannot be predicted

Proposed Answer: B

A. Incorrect: Water level would decrease not increase. Plausible because the output meter of the controller is higher than the upper meter. However, unlike the previous FWLC controllers, the upper meter in AUTO is not the signal from the Master controller. The top meter only indicates deviation between Auto and Manual when the controller is placed in the BAL-AUTO or BAL-MAN positions. In Auto or Manual the deviation meter is "centered" at the zero position.

B. Correct: Water level would decrease. The controller was placed in auto at 10% power. The procedure first opens the valve in manual and then places the controller in auto when level stabilizes.

Since the manual setting is set to control at 12% power, the valve will close to this position if the controller is placed in manual. Water level will therefore decrease.

C. Incorrect: Water level would decrease. Plausible if the operator believes that the deviation meter, the upper meter, indicates the current deviation between the manual setting and the auto setting. This is only true if the controller is in the BAL-AUTO or BAL-MAN positions. Other PNPS controllers are self-balancing and the controller is shifted from Auto to Manual without balancing (RBCCW and TBCCW controllers).

D. Incorrect: Water level would decrease. Plausible because the deviation meter (top meter) only indicates deviation when the controller is placed in the BAL-AUTO or BAL-MAN positions. If the operator focuses on this fact, the operator will conclude that the response of the valve cannot be predicted. However power was raised from 10% to 30% so the manual setting is less than the current auto setting and the valve will close.

Technical Reference(s)	PNPS 2.2.82, page 7		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-04-10, FWLC, EO-13		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	264000	A1.01
	Importance Rating	3.0	

Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including: Lube oil temperature

Proposed Question: RO Question # 14

The Emergency Diesels automatically started following a loss of off-site power.

A malfunction associated with the EDG "A" lube oil cooler results in rising lube oil temperature.

Lube oil temperature will rise until ...

A. engine failure occurs.

B. the diesel generator locks out.

C. the diesel shutdown relay trips on high lube oil temperature.

D the diesel shutdown relay trips on high uppermost bearing temperature

Proposed Answer: A

A. Correct: Although there is a trip on high Lube Oil temperature this trip is bypassed during an automatic start. Following an emergency start all trips are bypassed except overspeed, DG Lockout, and emergency manual shutdowns. The DG lockout is only activated by overcurrent or high differential current.

B. Incorrect: Following an emergency start the lube oil high temperature trip is bypassed. Plausible in that the lockout still functions during an emergency start but the lockout is only activated by overcurrent or high differential current.

C. Incorrect: Following an emergency start all trips are bypassed except overspeed, DG Lockout, and emergency manual shutdowns.

D. Incorrect: Following an emergency start all trips are bypassed except overspeed and DG Lockout. Plausible in that although there is not uppermost bearing high temperature trip, the low pressure trip is measured at this point.

Technical Reference(s)	PNPS 2.2.8, section 4.4, section 7.4 ARP C3L-A1	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-09-06, EO-22		(As available)
Question Source:	Bank Modified Bank New	WTSI 13012	Modified for Pilgrim (Note changes or attach parent)
Question History:	Last NRC Exam:	2010	Nine Mile Point 2
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41 55.43		7

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	
	Topic and K/A #	239002	A2.05
	Importance Rating	3.2	

Ability to (a) predict the impacts of the following on the RELIEF/SAFETY VALVES ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low reactor pressure

Proposed Question: RO Question # 15

Following a high drywell pressure condition, Emergency Depressurization is performed in accordance with EOP-17. All four Safety Relief Valves were placed in the OPEN position.

Current plant status is:

- The Red OPEN light above each SRV control switch is ON
- Reactor pressure is now 30 psig and rising slowly
- Drywell pressure is 3.1 psig and lowering slowly
- Core Spray is maintaining RPV level between +12 and +45 inches

Which one of the following is correct regarding the:

(1) Status of the SRVs and

(2) The required action, if any, to control reactor pressure?

A. (1) The SRVs are OPEN
(2) Leave the SRV control switches in OPEN.

B. (1) The SRVs are OPEN
(2) Return the SRV control switches to AUTO and place Shutdown Cooling in service.

C. (1) The SRVs are CLOSED
(2) Return the SRV control switches to AUTO and place Shutdown Cooling in service.

D (1) The SRVs are CLOSED
(2) Leave the SRV control switches in OPEN.

Proposed Answer: D

A. Incorrect: The SRVs are closed not open. Plausible in that the red light open lights are illuminated. The red indicating light above each SRV switch only indicates that the SRV solenoid valve is energized which would occur when the switch is placed in the open position to perform the ED.

B. Incorrect: The SRVs are closed not open. Additionally, Shutdown cooling cannot be placed in service because drywell pressure is over 2.2 psig. Plausible in that EOP-01 would direct this action if shutdown cooling could be placed in service.

C. Incorrect: Shutdown cooling cannot be placed in service because drywell pressure is over 2.2 psig.

D. Correct. RPV pressure is used to overcome the main valve preload spring tension (set at ~ 50 psig) which in turn opens the SRV when operating in the manual mode. The valve will physically close when RPV pressure is less than the preload spring tension. The question stem provides a condition where all SRVs have been manually opened for ED. When RPV pressure lowers to where there is < 50 psid between the RPV and Torus, the SRVs closed.

The red indicating light above each SRV switch only indicates that the SRV solenoid valve is energized which would occur when the switch is placed in the open position to perform the ED.

Once pressure control is shifted to the SRVs for an ED, the SRVs remain in control until the EOPs are exited. The SRVs will reopen if pressure rises to 50 psi above torus pressure. As further stated in 5.3.35.2, once the SRV control switches are placed in the "OPEN" position for Emergency RPV Depressurization, they shall remain in the "OPEN" position until the EOPs are exited.

Technical Reference(s)	EOP-17 5.3.35.2, section 4.3, item [5]		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-09, EO-3		(As available)
Question Source:	Bank Modified Bank New	15754	Modified for Pilgrim (Note changes or attach parent)
Question History:	Last NRC Exam:	2011	Oyster 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 #	2	
	Group #	1	
	Topic and K/A #	300000	A2.01
	Importance Rating	2.9	

Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: Air dryer and filter malfunctions

Proposed Question: RO Question # 16

The plant is at rated conditions with the instrument air system configured as follows:

- Instrument Air Dryer X-160 is in service.
- Air Dryer X-105 is available but not in service.

Then, a failure with the on-line drying tower of X-160 causes desiccant to carry over to the X-160 after filter. Air pressure downstream of the Air Dryers begins to lower.

Assuming air header pressure lowers to 83 psig, which one of the following is correct regarding:

(1) any automatic action AND

(2) actions required to restore instrument air dryer operation?

A. (1) Standby air dryer train X-105 isolation valves open.
(2) Manually isolate the X-160 air dryer train.

B. (1) Standby air dryer train X-105 isolation valves open.
(2) Verify X-160 isolation valves automatically close when X-105 isolation valves open and pressure recovers.

C. (1) AO-4310, Instrument Air Dryer Bypass Valve, opens.
(2) Manually isolate the X-160 after filter and place X-105 dryer in service. Manually Close AO-4310.

D (1) AO-4310, Instrument Air Dryer Bypass Valve, opens
(2) Manually isolate the X-160 after filter and place X-105 dryer in service. Verify AO-4310 automatically closes when differential pressure across the X-160 train lowers.

Proposed Answer: C

A. Incorrect: Dryer trains do not automatically shift. Plausible in that the towers within the train do shift under certain conditions.

B. Incorrect: Dryer trains do not automatically shift.

C. Correct: Normally closed AO-4310 opens at 84 psig and lowering to bypass the dryer trains in order to recover pressure. IAW with PNPS 5.3.8, the X-160 filter must be removed from service after verifying AO-4310 open. Then the X-105 dryer must be placed in service and AO-4310 manually closed to route instrument air through the dryer.

D. Incorrect: AO-4310 does not automatically re-close.

Technical Reference(s)	PNPS 5.3.8 section 2.0, Subsequent actions [10] and [20]		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-NL-03-09-01, EO-10j		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	205000	A3.01
	Importance Rating	3.2	

Ability to monitor automatic operations of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) including: Valve operation

Proposed Question: RO Question # 17

Given the following:

- The plant is in cold shutdown .
- "A" RHR pump is running in Shutdown Cooling (SDC) Mode.
- RPV Level is +35 inches

Then the following occur:

- All off-site power is lost and ALL 4160 VAC buses de-energize
- Both EDGs start and repower A5 and A6
- RPV level rises to +40 inches and stabilizes

Which one of the following correctly describes the status of the Shutdown Cooling Lineup two (2) minutes later?

	MO-1001-50 and 47, SDC Inboard and Outboard Isol Vlvs,	**MO-1001-29A, LPCI Injection Vlv #1**
A.	Open	Open
B.	Closed	Open
C.	Open	Closed
D	Closed	Closed

Proposed Answer: D

A. Incorrect - All three valves close when the logic is de-energized. Plausible that no valves go closed because the Group 3 logic relaying is powered from DC and never loses power. But because the low water level (and other) sensors are de-energized, the DC powered logic thinks level is low and isolates.

B. Incorrect - All three valves close when the logic is de-energized. Plausible in that MO-1001-29A does not go closed on all Group III isolations.

C. Incorrect– All three valves close when the logic is de-energized.

D. Correct - Even though level didn't reach the Group 3 isolation setpoint, a Group 3 isolation occurred because the Group 3 sensing logic is powered from RPS. When off-site power was lost, RPS de-energized causing the Group 3 isolation. All three valves close when the logic is de-energized.

Technical Reference(s)	2.2.125.1, section 4.3 and PCIS Reference Text, Figures 2a, 3a, and 3b	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-09-01, EO-24.a	(As available)
Question Source:	Bank PNPS Bank	
	Modified Bank	(Note changes or attach parent)
	New	
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	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	
	Topic and K/A #	203000	A3.06
	Importance Rating	3.7	

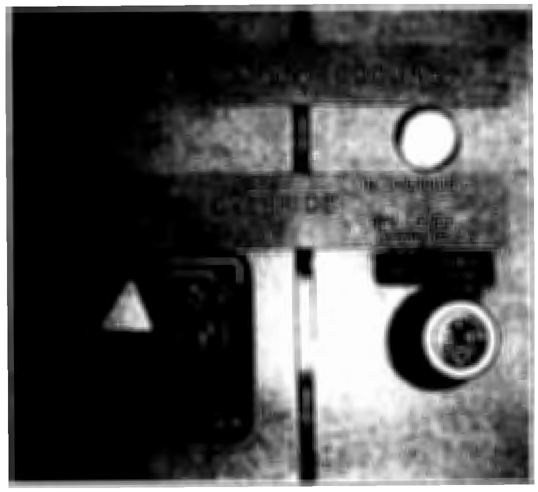
Ability to monitor automatic operations of the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) including: Indicating lights and alarms

K/A Justification: This question tests the candidate's knowledge of those conditions that automatically align the RHR system for dedicated injection after containment cooling has been established.

Proposed Question: RO Question # 18

The reactor is scrammed following a high drywell pressure condition. Plant conditions are as follows:

- Both loops of RHR have been placed in torus cooling and torus sprays.
- RPV level is being controlled between +12 and +45 inches using HPCI
- Reactor pressure is 500 psig and lowering.
- Drywell pressure 4 psig and lowering



Then, HPCI trips and RPV level begins to lower.

The white light shown to the right will remain illuminated UNTIL ...

A. RPV level lowers to -46 inches

B. RPV level lowers to -150 inches

C. Reactor pressure lowers to 400 psig

D Drywell pressure lowers to < 2.2 psig

Proposed Answer: B

A. Incorrect: RPV level must lower to -150 inches. Plausible because -46 inches is a LPCI loop selection signal.

B. Correct: Drywell pressure >2.2 psig is a LPCI Initiation. The initiation signal interlocks the containment cooling valves closed. This interlock can be overridden by placing LPCI Override switch in the Manual Override position as shown in the picture. The white light will illuminate when the interlock is overridden.

However if RPV level lowers to -150 inches then the override is removed and the white light extinguishes. This is to ensure that all flow is aligned to the vessel. This can also be overridden by using the keylock switch if desired.

C. Incorrect: Not until level lowers to -150". Plausible in that at 400 psig, the injection valves will open.

D. Incorrect: Not until level lowers to -150". Plausible in that high drywell pressure is a LPCI initiation signal. However this signal remains sealed in until the operator manually resets the initiation signal.

Technical Reference(s)	PNPS 2.2.19, section 4.2.4, items [2], and [5] RHR reference text, page 73		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-09-01, RHR, EO-14		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	218000	A4.10
	Importance Rating	3.8	

ADS: Ability to manually operate and/or monitor in the control room: Lights and alarms

Proposed Question: RO Question # 19

The reactor is at rated conditions when a large RWCU leak outside the drywell occurs.

- The leak cannot be isolated.
- No sources of high pressure injection are available.
- RPV pressure is 900 psig and steady.
- RPV level is -35 inches and lowering

Regarding the ADS affiliated alarms listed below, what is the sequence at which these alarms will annunciate if RPV level continues to lower? Assume NO operator action is taken.

Alarm 1: 2 MIN TIMER INITIATED, C903L-C1
Alarm 2: 11 MIN TIMER INITIATED, C903L-D1
Alarm 3: DRYWELL PRESSURE HI, C903L-E1
Alarm 4: RHR/CS PUMP RUNNING, C903L-F1
Alarm 5: RELIEF/SFTY VALVE OPEN, C903L-B2

A. 1 and 2 simultaneously
Then 4
Then 5

B. 2 first
Then 1 and 4 almost simultaneously
Then 5

C. 2 first
Then 1, 3 and 4 almost simultaneously
Then 5

D 1 first
Then 2
Then 4 and 5 almost simultaneously

Proposed Answer: B

A. Incorrect: The first annunciator will be the 11 minute timer alarm. Plausible in that this sequence of alarms would be correct if there was a high drywell pressure condition.

B. Correct: The two minute timer will activate if level is <-46" and drywell pressure is > 2.2 psig. However, the leak outside the drywell will generate a low-low level signal of -46 inches but not a high drywell pressure condition.

The 11 minute timer will start at -46 inches and if this condition is sustained for 11 minutes, the high drywell pressure requirement for the two minute timer to start is bypassed. The 2 minute timer will then start. Also, when the 11 minute timer times out, an auto start signal is sent to the low pressure ECCS. The RHR/CS pump running alarm will annunciate as soon as one pump's discharge pressure is > than the setpoint. After the two minute timer times out the ADS valves will open.

C. Incorrect: The high drywell pressure alarm will not annunciate. Plausible if the operator understands that the 11 minute timer takes the place of the high drywell pressure requirement and will trigger that alarm at the conclusion of the 11 minutes.

D. Incorrect: The first annunciator will be the 11 minute timer alarm. Plausible if the operator remembers that a -46" signal inputs into the 2 minute timer and that the 11 minute timer must also time out if there is no drywell pressure signal. The operator may then conclude that after the 2 minute timer times out, the 11 minute timer starts which will start the ECCS pumps completing the ADS logic.

Technical Reference(s)	PNPS 2.2.23, section 4.2 ADS Reference Text, Figures 6 and 7	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-09-05, ADS LP, EO 11		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	400000	A4.01
	Importance Rating	3.1	

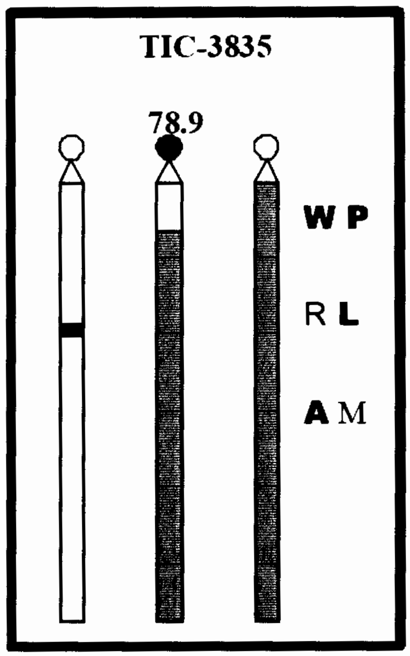
Ability to manually operate and/or monitor in the control room: CCW indications and control

Proposed Question: RO Question # 20

While operating at full power temperatures begin to rise on components cooled by RBCCW loop 'B'.

The loop 'B' temperature controller indicates as shown in the picture at the right.

Based on these conditions which one of the following is consistent with these indications?



A. The controller output has failed upscale.

B. The temperature control valve has failed open locally.

C. The temperature control valve has failed closed locally.

D There is insufficient Salt Service Water flow to the heat exchanger.

Proposed Answer: A

A. Correct: The RBCCW controller regulates a bypass valve around the heat exchanger. The higher the output (right bar chart), the more open the bypass flow and the less cooling provided by the heat exchanger.

The setpoint is represented by the left bar chart, while the actual temperature is the middle bar chart. Since the actual loop temperature is higher than the setpoint the controller should be calling for more cooling, i.e., less bypass flow. Instead it is calling for maximum bypass flow which will raise the temperature further.

Therefore the automatic feature of the controller has failed. The appropriate action is to place the controller in manual and lower the output and thereby reduce the amount of bypass flow.

B. Incorrect: The controller output has failed. Plausible in that If the temperature control valve failed open, temperature would rise. However the controller would sense the temperature rise and its output would be low not high as it attempted to lower temperature.

C. Incorrect: The controller output has failed. If the temperature control valve had failed closed there would be maximum cooling to the heat exchanger and temperature would lower. Plausible in that the controller output would increase as it attempted to reduce the amount of cooling.

D. Incorrect: The controller output has failed. Plausible in that insufficient flow would cause the rise in temperature but the controller output would be low as it attempted to close the bypass valve.

Technical Reference(s)	PNPS 2.2.30, Attachment 6 RBCCW Ref Text, Fig 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-02-06, RBCCW LP, EO-15e		(As available)
Question Source:	Bank	PNPS Bank 12323	
	Modified Bank		(Note changes or attach parent)
	New		
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		4

55.43
Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	Topic and K/A #	223002	2.1.23
	Importance Rating	4.3	

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation. PCIS/Nuclear Steam Supply Shutoff

Proposed Question: RO Question # 21

The plant is at rated power. Drywell venting using Standby Gas Treatment Train "A" is in progress to maintain primary containment differential pressure.

Then,

- A feedwater transient results in a reactor scram
- A Reactor Building Isolation System (RBIS) actuation occurs due to low vessel level
- RPV level has now been restored to +25 inches.

Ten (10) minutes later the RBIS is in the process of being reset.

IAW PNPS 2.2.125.1, RESET OF PRIMARY AND SECONDARY CONTAINMENT ISOLATIONS:

Which of the following manual actions are required to be taken PRIOR to resetting the RBIS trip logic using the RBIS TRIP RESET keylock reset switches at Panel C7?

Action 1: Place all control switches for the Reactor Building Supply and Exhaust fans in the "OFF" position.

Action 2: Place all control switches for the Reactor Building Supply and Exhaust fans in the "STANDBY" position.

Action 3: Place all control switches for the Reactor Building Supply and Exhaust isolation dampers control switches in the "AUTO" position.

Action 4: Place all control switches for the Reactor Building Supply and Exhaust Isolation Dampers in the "CLOSE" position.

Action 5: Manually start Standby Gas Treatment Train "B"

Action 6: RESET the Group II isolation at Panel C905 using the PCIS GROUP 2, 3, 6 ISOL RESET switch at Panel C905

A. Actions 2 and 3 only

B. Actions 2, 3 and 6, only

C. Actions 1, 4 and 5 only

D Actions 1, 4, 5, and 6 only

Proposed Answer: D

A. Incorrect: Supply and exhaust fans are placed in OFF, dampers are placed in CLOSE. Plausible if the candidate thinks the system should be aligned for normal operation standby lineup prior to resetting. Additionally SBGT "B" will need to be started and the Group II isolation reset. Not resetting the Group II isolation is plausible if the candidate does not understand that the PCIS logic status inputs into the RBIS logic and recalls that not all RBIS isolation signals are PCIS signals.

B. Incorrect: Supply and exhaust fans are placed in OFF, dampers are placed in CLOSE. Additionally SBGT "B" will need to be started.

C. Incorrect: The Group II isolation will need to be reset.

D. Correct: As discussed via the procedure caution in Attachment 8 of PNPS 2.2.125.1, the NRC requires that all safety related equipment receiving a containment isolation signal remain in the safe position when containment isolation logic is reset. If the below listed components are not aligned as required below, the components will reposition when the isolation logic is reset.

- Rx Building supply and exhaust fans
- Rx Building isolation dampers
- SBGT trains

Therefore Attachment 8 requires:

- That all control switches for the Reactor Building Supply and Exhaust fans be placed in the "OFF" position (action 1)
- That all control switches for the Reactor Building Supply and Exhaust isolation dampers be placed in the "CLOSE" position (action 4)
- That both standby gas treatment fans be running. Both trains received an auto start signal. However train "B" would have shut down after 60 seconds. Therefore train "B" would need to be started. This is accomplished by re-positioning train "B" dampers (action 5)
- The low RPV level condition would also generate a Group II isolation in addition to the RBIS. The Group II isolation must therefore also be reset (action 6)

Technical Reference(s)	PNPS 2.2.125.1, Attachment 8	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RQ-02-08-10, PCIS, EO-12i		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	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	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	
	Topic and K/A #	212000	2.1.7
	Importance Rating	4.4	

Conduct of Operations: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation. RPS

Proposed Question: RO Question # 22

During an event initiated by a load reject and loss of off-site power, EOP-02 is being executed due to a hydraulic lock on the scram discharge volumes.

The electric plant responded as designed to the transient and" EDGs "A" and "B are supplying buses A5 and A6. Additional plant conditions are as follows:

- Rx Level is being controlled between -125 to -150 inches
- Rx Pressure is being controlled between 400 and 500 psig
- RPS and ARI have been bypassed

Under these conditions which one of the following describes the need for restoring RPS and CRD in order to perform repeated scrams?

A. BOTH RPS buses must be restored
At least ONE CRD pump must be restored

B. At least ONE RPS bus must be restored
At least ONE CRD pump must be restored

C. At least ONE RPS bus must be restored
A CRD pump is NOT required

D BOTH RPS buses must be restored
A CRD pump is NOT required

Proposed Answer: B

A. Incorrect: Only one RPS bus is required to reset the scram if the RPS trip logic is bypassed. Plausible in that normally, both RPS buses are required due to the arrangement of the SDIV high level scram logic.

B. Correct: Only one RPS bus is required if the RPS has been bypassed. At least one CRD pump is required because RPV pressure is < 800 psig.

C. Incorrect: One CRD pump is required. Plausible in that this would not be true if pressure was > 800 psig.

D. Incorrect: Only one RPS bus is required and one CRD pump is required. See above.

Technical Reference(s)	PNPS 5.3.23, section 3.3	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:	None

Learning Objective:	O-RO-02-07-07, RPS LP, EO-15	(As available)

Question Source:	Bank	LOR Bank 341	
	Modified Bank		(Note changes or attach parent)
	New		

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	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	
	Topic and K/A #	211000	A4.06
	Importance Rating	3.9	

Ability to manually operate and/or monitor in the control room: RWCU system isolation

Proposed Question: RO Question # 23

The plant was at rated power when an event occurred resulting in an ATWS.

The SLC ACTUATE switch has just been placed in the "SYS B" position.

ONE MINUTE later, which of the following is correct regarding the status of the following correct Reactor Water Cleanup (RWCU) valves?

	MO-1201-2, INBD ISOL VLV	MO-1201-5, OUTBD ISOL VLV	MO-1201-80, RETURN ISOL VLV
A.	Open	Closed	Closed
B.	Closed	Open	Open
C.	Closed	Closed	Closed
D	Closed	Closed	Open

Proposed Answer: C

A. Incorrect: All three valves close. Plausible if the operator believes that going to Sys “B” only actuates the outboard isolation logic.

B. Incorrect: All three valves close. Plausible if the operator believes that going to Sys “B” only actuates the inboard isolation logic.

C. Correct: Placing the SLC Actuate Switch in either position will actuate both isolation logics.

D. Incorrect: All three valves will close. Plausible if the operator does not think the return valve closes.

Technical Reference(s)	PNPS 2.2.83, section 4.4, item [2]	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None

Learning Objective:	O-RO-02-06-06, SBLC, EO-17		(As available)
Question Source:	Bank Modified Bank New	WTSI 15756	Modified for Pilgrim (Note changes or attach parent)
Question History:	Last NRC Exam:	2011	Oyster Creek

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

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	
	Topic and K/A #	217000	A3.06
	Importance Rating	3.5	

Ability to monitor automatic operations of the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) including: Lights and alarms

Proposed Question: RO Question # 24

Following a loss of off-site power the following occur:

- RCIC has received an automatic start signal
- As RCIC turbine speed is increasing, the following alarms are received:
    - Alarm C904L-A1, RCIC Isolated
    - Alarm C904L-B1, Steam Line High Flow

Which one of the following indicates that RCIC FAILED to respond as designed?

The position indicating lights for ...

A. MO-1301-61, RCIC Turbine Steam Inlet Valve, indicate the valve is open

B. MO-1301-60, RCIC Pump Minimum Flow Valve, indicate the valve is open

C. MO-1301-49, RCIC Pump Discharge Injection Valve #2, indicate the valve is open

D MO-1301-22, RCIC Pump Condensate Storage Tank Suction Valve, indicate the valve is open

Proposed Answer: B

A. Incorrect: This is a correct indication. MO-1301-61 opened as a result of the initiation signal. RCIC has also received an isolation signal. MO-1301-61 does not close on an isolation signal. Plausible in that it does close on high RPV water level.

B. Correct: The High Steam Flow alarm is the result of a RCIC Isolation signal. MO-1301-60 should automatically close when the TT&T valve closes to prevent draining the CST to the torus. The TT&T valve closes via the isolation signal.

C. Incorrect: This is a correct indication. MO-1301-49 opened as a result of the initiation signal. MO-1301-49 does not close on an isolation signal.

D. Incorrect: This is a correct indication. RCIC has received an isolation signal. MO-1301-22 does not close on an isolation signal.

Technical Reference(s)	ARP for alarm C904L-B1	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-09-04, RCIC LP, EO-11	(As available)
Question Source:	Bank 11385 Modified Bank New	Modified for Pilgrim (Note changes or attach parent)
Question History:	Last NRC Exam: 2010	Nine Mile Point 2
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	 X
10 CFR Part 55 Content:	55.41 55.43	7

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	
	Topic and K/A #	223002	K3.06
	Importance Rating	2.8	

Knowledge of the effect that a loss or malfunction of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF will have on the following: Turbine building radiation

Proposed Question: RO Question # 25

The plant is at 100% power, and operating with a minor fuel defect when alarm CRD/DRYWELL/MISC AREAS TEMP HI, C903R-A1, is received.

The KAYE Computer indicates that Turbine Building Area Temperatures for the Turbine Building Basement /Condenser Compartment are in alarm and rising

Two minutes later, alarm Steam Leakage Area Temp HI, C904L-A6 is received. The BOP operator reports that the Turbine Basement Exhaust is reading 170 °F and rising.

Which one of the following is correct regarding :

(1) The response of the Primary Containment Isolation System (PCIS) AND

(2) Those radiation monitors that are impacted by this event?

A. (1) The PCIS has failed to respond as designed.
(2) Area Radiation Monitor #2, Condensate Pump Stairway
Main Stack Radiation Monitors.

B. (1) The PCIS has failed to respond as designed.
(2) Area Radiation Monitor #1, Feedwater Heaters Stairwell
Reactor Building Vent Radiation Monitors.

C. (1) The PCIS isolation setpoints have NOT been exceeded and the system is responding as designed.
(2) Area Radiation Monitor #1, Feedwater Heaters Stairwell
Reactor Building Vent Radiation Monitors.

D (1) The PCIS isolation setpoints have NOT been exceeded and the system is responding as designed.
(2) Area Radiation Monitor #2, Condensate Pump Stairway
Main Stack Radiation Monitors.

Proposed Answer: B

A. Incorrect: Area Radiation Monitor #2 would not be expected to respond to this event as it is outside the Turbine Building Basement which is also known as the condenser compartment. Additionally the Main Stack Rad monitors would not be expected to respond. The Turbine Building basement ventilation exhausts to the Reactor Building Vent. The Main Stack only receives the discharge from Standby Gas, Offgas, Primary containment atmosphere control (in purge or inerting mode), and Control room HVAC.

B. Correct: The indications provided are indicative of a high energy line break in the condenser compartment (extraction steam, feedwater etc.) The Group I high temperature isolation is triggered by either high temperature in the Steam Tunnel or by Turbine Basement Exhaust temperature. The setpoint for the turbine basement exhaust is 155 °F. Therefore the Group I isolation has failed.

The symptoms provided are indicative of a break in the condenser compartment. The only turbine building ARM in the condenser compartment is ARM #1, Feedwater Heaters Stairwell. Additionally the turbine basement exhausts to the Reactor Building Vent. Therefore the Reactor Building Vent rad monitors would respond to the increase in radiation.

C. Incorrect: A Group I isolation should have occurred due to Turbine Building Basement high temperature > 155 degrees. Plausible if the operator does not recall that the Main Steam High Temp isolation is triggered by the Turbine Basement Exhaust in addition to Main Steam tunnel temp. Or the operator may confuse the Turbine Basement Exhaust trip point with the 175 degrees setpoint for the Main Steam Tunnel.

D. Incorrect: A Group I isolation should have occurred due to Turbine Building Basement high temperature > 155 degrees. Additionally, ARM # 2 and the Main Stack Rad monitors are not expected to respond.

Technical Reference(s)	ARPs C905L - A1, MAIN STM LINE AREA TEMP HI PNPS 2.2.125, Attachment 3 Rx Build Ventilation Ref Text, figure 1.		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-03-03, PRM Lesson plan, EO-4		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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 #	2	
	Group #	1	
	Topic and K/A #	209001	K2.01
	Importance Rating	3.0	

Knowledge of electrical power supplies to the following: Pump power

Proposed Question: RO Question # 26

Which one of the following DC loses would prevent Core Spray Pump "A" from automatically starting following a high drywell pressure condition?

Loss of ...

A. D-4

B. D-5

C. D-6

D D-7

Proposed Answer: A

A. Correct: Core Spray pump "A" is supplied from 4160 VAC bus A5. 125 VDC panel D-4 supplies breaker control power for the A5 breakers. D-4 also supplies the logic power necessary to start the pump.

B. Incorrect: Incorrect: A loss of D-4 would prevent the pump from starting. D-5 supplies logic power and breaker control for Core Spray "B"

C. Incorrect: A loss of D-4 would prevent the pump from starting. D-6 supplies mainly turbine and generator protective circuits and B-6 transfer logic.

D. Incorrect: A loss of D-4 would prevent the pump from starting. D-7 supplies RCIC.

Technical Reference(s)	Core Spray Ref Text Table 1	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-01-02, 125 VDC LP, EO-13a O-RO-02-09-02, CS LP, EO-7f	(As available)
Question Source:	Bank	

	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	219000	K1.06
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause- effect relationships between RHR/LPCI: TORUS/SUPPRESSION POOL COOLING MODE and the following: Keep fill system

Proposed Question: RO Question # 27

The plant was at rated conditions when a major LOCA occurred.

The one available Torus Cooling Loop is placed in Maximized Torus Cooling IAW PNPS 2.2.19.5, RHR MODES OF OPERATION FOR TRANSIENTS.

The pump in torus cooling is the only low pressure ECCS pump currently running.

When Torus Water temperature exceeds 130°F, MO-1001-36A, Torus Cooling Valve, is opened fully to achieve loop flow at the maximum RHR Pump capacity.

Shortly thereafter, alarm "RHR/CS PUMP RUNNING" (C903-F1) CLEARS.

Which one of the following is correct regarding the significance of this alarm clearing and any required action?

A. Pump damage has occurred. Shift the running RHR pump to the pump in standby.

B. The pump is running out. Throttle MO-1001-36A in the close direction until the alarm clears.

C. Keepfill pressure is insufficient to keep the piping full. Start the 2nd RHR pump to keep the piping full.

D Discharge pressure is now lower than keepfill pressure and CST water will begin to fill the torus. Monitor torus level.

Proposed Answer: D

A. Incorrect: This is an expected occurrence for this lineup. The pump should not be secured based on this indication alone.

B. Incorrect: The pump is not running out. Pump runout occurs at 5600 gpm. The maximum flow that one RHR Pump will deliver to the heat exchanger is less than 5600 GPM with the Torus Cooling or LPCI injection valves full open.

C. Incorrect: This is an expected occurrence for this lineup. Starting the 2nd pump would require opening the heat exchanger bypass valve which would reduce the heat removal rate.

D. Correct: This is an expected occurrence for the lineup. Normally when in torus cooling, flow is limited to ~ 5000 gpm. Fully opening MO-1001-36A will increase flow further. As discussed in the caution of PNPS 2.2.19.5, Attachment 3, page 3 of 3, Going above 5100 GPM while in Torus Cooling single pump operation will lower RHR Pump discharge pressure lower than keepfill pressure and CST water will begin to fill the Torus through the keepfill lines. This condition can be initially noticed if alarm "RHR/CS PUMP RUNNING" (C903-F1) clears. Step 12 of this attachment, directs that if necessary the keepfill block valve can be closed.

Technical Reference(s)	PNPS 2.2.19.5, page 11, Attachment 3, steps 10 through 12	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-09-01, RHR, EO-15h	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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	
	Topic and K/A #	201001	K2.03
	Importance Rating	3.5	

Knowledge of electrical power supplies to the following: Backup SCRAM valve solenoids

Proposed Question: RO Question # 28

Concerning the Backup Scram Valves:

Which one of the following is correct regarding

(1) The power supply to the Backup Scram Valve solenoids AND

(2) Whether a loss of this power supply would result in a scram?

A. (1) 120 VAC RPS power
(2) A reactor scram would NOT occur

B. (1) 120 VAC RPS power
(2) A reactor scram WOULD occur

C. (1) 125 VDC power
(2) A reactor scram would NOT occur

D (1) 125 VDC power
(2) A reactor scram WOULD occur

Proposed Answer: C

A. Incorrect: The solenoids are powered from 125 VDC. Plausible in that the scram pilot valves solenoids are powered from 120 VAC RPS power.

B. Incorrect: The solenoids are powered from 125 VDC and a scram would not occur. Plausible in that the scram pilot valves solenoids are powered from 120 VAC RPS power and that a complete loss of RPS power would cause a scram.

C. Correct: 125 VDC bus D-4 supplies solenoid "A" and D-5 supplies solenoid "B". These solenoids are normally de-energized and energize to vent off the scram air header. Therefore a loss of power would not cause a scram.

D. Incorrect: A scram would not occur. These solenoids are normally de-energized. A loss of power will not cause these solenoids to change position.

Technical Reference(s)	CRDH Ref text, pages 24, 25 and figure 9	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-06-11, CRDH LP EO-12i	(As available)

Question Source:	Bank	WTSI 12627	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Hope Creek

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	
	Topic and K/A #	201002	K3.03
	Importance Rating	2.9	

Knowledge of the effect that a loss or malfunction of the REACTOR MANUAL CONTROL SYSTEM will have on following: Ability to process rod block signals

Proposed Question: RO Question # 29

Reactor power is 2% with the Rx. Mode Switch in STARTUP during a reactor plant startup. Additional information is as follows:

- Control rod 26-27, currently at position 12, is scheduled to be withdrawn to position 24.
- When the operator attempts to withdraw control rod 26-27 one notch, the Reactor Manual Control Timer fails such that the withdraw "bus" remains continuously energized

Which ONE of the following will terminate the rod withdraw transient?

A. Rod Worth Minimizer will insert a withdraw block when the rod withdraws past position 24.

B. Rod Block Monitor will insert a withdraw block when local power around the rod increases by 20%.

C. Reactor Manual Control will de-select the control rod and block further rod selection.

D APRMs will insert a withdraw block when power increases to 12%.

Proposed Answer: C

A. Incorrect: The RMCS timer select block will stop the rod motion before the rod reaches notch 24.

B. Incorrect: The RBM is bypassed at this power level.

C. Correct: The normal notch out withdraw portion of the sequence takes 1.5 seconds. If a withdraw signal is sent to directional control valves for more than 2 seconds, an auxiliary timer times out. When the auxiliary timer times out, it generates a select block. This deselects the selected rod and prevents further rod selection until the block has been cleared. This prevents a faulty master timer from causing an uncontrolled continuous withdrawal signal.

D. Incorrect: The RMCS timer select block will stop the rod motion before the power increases to 12%.

Technical Reference(s)	RMCS Reference Text page 14. ARP 905L-C4	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-08, EO-10a (As available)

Question Source:	Bank	TADS 11800	
	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	6
	55.43	

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Topic and K/A #	215001	K4.01
	Importance Rating	3.4	

Knowledge of TRAVERSING IN-CORE PROBE design feature(s) and/or interlocks which provide for the following: Primary containment isolation: Mark-I&II(Not-BWR1)

Proposed Question: RO Question # 30

The plant was operating at rated conditions with a TIP trace in progress, when a transient causes reactor water level to lower to -10 inches before recovering to +30 inches.

Which ONE of the following correctly describes how this will affect, if at all, the operation of the TIP system?

A. The TIP will continue WITHOUT interruption.

B. The TIP will stop AND the shear valve will fire.

C. The TIP withdraws to the Parked position AND the ball valve closes.

D The TIP withdraws to the In-Shield position AND the ball valve closes.

Proposed Answer: D

A. Incorrect: The TIP withdraws to the In-Shield position AND the ball valve closes. Plausible if the operator does not recognize that an automatic withdrawal signal is generated when level lowers to < +12 inches.

B. Incorrect: The TIP withdraws to the In-Shield position AND the ball valve closes. The shear valve will not automatically fire.

C. Incorrect: The TIP withdraws to the In-Shield position AND the ball valve closes. Plausible in that following a scan the TIP automatically withdraws to the Parked position.

D. Correct: A group 2 isolation signal was generated when level lowered to +12 inches. This caused the TIP to automatically withdraw to the in-shield position and the ball valve closes.

Technical Reference(s)	PNPS 2.2.69, pages 11 and 12	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None

Learning Objective:	O-RO-02-07-08 Objective 56e		(As available)
Question Source:	Bank Modified Bank New	10896	Modified for Pilgrim (Note changes or attach parent)
Question History:	Last NRC Exam:	2009	Dresden

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

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	
	Topic and K/A #	234000	K5.01
	Importance Rating	2.9	

Knowledge of the operational implications of the following concepts as they apply to FUEL HANDLING EQUIPMENT : Crane/hoist operation

Proposed Question: RO Question # 31

The plant is refueling. The Reactor Mode Switch is in REFUEL

Which ONE of the following would cause a Control Rod Withdrawal Block?

A. Fuel Grapple loaded AND
the Refueling Platform over the spent fuel pool

B. Refuel Platform monorail mounted hoist loaded AND
the Refueling Platform over the core

C. One control rod at position 48 AND
the Refueling Platform over the core AND
Hoists and Grapple all unloaded

D One control rod at position 48 AND
the Refuel Platform monorail mounted hoist loaded AND
the Refueling Platform over the spent fuel pool

Proposed Answer: B

A. Incorrect: This statement is false because the Refueling Platform is over the Spent Fuel Pool.

B. Correct: Refuel Platform monorail mounted hoist loaded AND the Refueling Platform over the core AND the Reactor Mode Switch in Refuel will cause a Control Rod Withdrawal Block

C. Incorrect: The rod block would not occur until a second rod was selected.

D. Incorrect: The rod block would not occur until a second rod was selected, or the bridge was near or over the core.

Technical Reference(s)	2.2.75, Attachment 4, sheet 1 of 3	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-06-04-01, EO-9a		(As available)
Question Source:	Bank Modified Bank New	X	See comment (Note changes or attach parent)
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 55.43		6

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Topic and K/A #	272000	K6.02
	Importance Rating	2.5	

Knowledge of the effect that a loss or malfunction of the following will have on the RADIATION MONITORING SYSTEM : D.C. power

Proposed Question: RO Question # 32

A complete loss of 24 VDC has occurred. Which one of the following will now occur?

A. The Mechanical Vacuum pump will trip if running.

B. Reactor Building ventilation will isolate and Standby Gas will start.

C. The Off-Gas isolation 13 minute timer will initiate if the Off-gas isolation keylock selector switch is selected to Pre-Treat Position 2.

D The Off-Gas isolation 13 minute timer will initiate if the Off-gas isolation keylock selector switch is selected to Post-Treat Position 1.

Proposed Answer: D

A. Incorrect: The Main Steam line rad monitors trigger the vacuum pump trip. However these monitors are powered from RPS power.

B. Incorrect: The refuel floor vent exhaust monitors trigger the Reactor Building ventilation isolation and Standby Gas start. However these monitors are powered from both RPS power and 120 VAC Vital Bus Y-2

C. Incorrect: When the Off-gas isolation keylock selector switch is selected to Position 2, RM1705-3A and B input into the Offgas isolation circuit. However these monitors are powered from RPS power.

D. Correct: Post Treat Rad Monitors RM1705-5A and B are powered from the 24VDC system. When the Off-gas isolation keylock selector switch is selected to Position 1 these monitors input into the off gas isolation logic. This logic will actuate if both of these rad monitors have downscale trips. The downscale trips will be actuated on loss of power.

Technical Reference(s)	ARP CP600R-C6 PRM Ref. Text, page 33	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-04-03 #13		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	X	
	Comprehension or Analysis		
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 #	2	
	Group #	2	
	Topic and K/A #	201006	A1.01
	Importance Rating	3.2	

Ability to predict and/or monitor changes in parameters associated with operating the ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) controls including: Rod position: P-Spec(Not-BWR6)

Proposed Question: RO Question # 33

Control rods are being withdrawn to heatup and pressurize the reactor plant.

Regarding Control Rod 10-27:

- Control rod 10-27 is in RWM step 46
- The rod is being moved from position 18 to 24
- The reed switch for position 20 is known to be failed
- A Substitute Rod position has been entered for position 20

Which one of the following is correct?

The substituted position ...

A. will be automatically removed when the rod is withdrawn to position 22

B. will be automatically removed when the rod reaches its withdraw limit of position 24

C. must be manually removed when the rod is moved to a position with an operable reed switch at a control room EPIC workstation

D must be manually removed when the rod is moved to a position with an operable reed switch by re-initializing the system via the System Initialize button on the RWM display.

Proposed Answer: C

A. Incorrect: The substituted positon must be manually removed. Plausible in that the previous RWM used to automatically perform this function.

B. Incorrect: The substituted positon must be manually removed.

C. Correct: The substituted positon must be manually removed IAW Attachment 3 of PNPS 2.2.90. This is done at any control room EPIC workstation.

D. Incorrect: Re-initializing the system will remove the substituted rod position but is not directed per 2.2.90 attachment 3.

Technical Reference(s)	Attachment 3 of PNPS 2.2.90	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-07-06, TO	(As available)

Question Source:	Bank	LOR bank #143	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	Not Used	

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

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Topic and K/A #	202002	A2.01
	Importance Rating	3.4	

Ability to (a) predict the impacts of the following on the RECIRCULATION FLOW CONTROL SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Recirculation pump trip

Proposed Question: RO Question # 34

The plant was at 100% power when Recirc Pump “A” tripped due to a motor generator fault.

Following the pump trip, MO-202-5A, PUMP DISCH VLV, was manually closed. NO other actions have been taken.

Based on the above, the Right Bar Chart on the Recirc Pump “A” controller is currently indicating ___ (1) ___ (26% / 53%) AND

MO-202-5A is to be reopened when a minimum of ___ (2) ___ (3 minutes/ 5 minutes) have elapsed in order to maintain idle loop temperature.

A. (1) 26%
(2) 3 minutes

B. (1) 26%
(2) 5 minutes

C. (1) 53%
(2) 3 minutes

D (1) 53%
(2) 5 minutes

Proposed Answer: D

A. Incorrect: The output should be 53%. Plausible because the #1 limiter is activated when the discharge valve is closed. This limiter normally limits the output to 26%. However following the pump trip the startup signal generator overrides all other signals and set's the output to 53%. Additionally the discharge valve is not reopened until at least 5 minutes have elapsed.

B. Incorrect: The output should be 53%.

C. Incorrect: The discharge valve is not opened until at least 5 minutes have elapsed.

D. Correct: Following a pump trip the Recirc Flow Control system's startup signal generator adjusts the controller output to 53% to provide sufficient break away torque when the pump is started. The signal is inserted when the field breaker opens following the pump trip. Additionally, MO-202-5A is re-opened after at least 5 minutes have elapsed to keep the idle loop warm.

Technical Reference(s)	RFC Ref Text Figure 2 and page 11 PNPS 2.4.17, subsequent action [5]	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-10, EO-5 (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content:	55.41	6
	55.43	

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Topic and K/A #	256000	A3.08
	Importance Rating	3.1	

Ability to monitor automatic operations of the REACTOR CONDENSATE SYSTEM including: Feedwater temperature

Proposed Question: RO Question # 35

The plant is operating at rated conditions when a tube leak occurs in 3rd point feedwater heater, E-103A. The 3rd point heater level control valve cannot maintain level and heater level continues to rise.

Assuming the level continues to rise until the heater is full and NO operator action is taken, which one of the following is correct regarding any

(1) Automatic response of heater level control AND

(2) Subsequent changes in final feedwater temperature going into the reactor?

A. (1) 3rd point heater dump valve opens
3rd point heater Bleeder Trip Valve closes and Spill valve opens
(2) Feedwater temperature rises

B. (1) 2nd point heater drain valve closes
3rd point heater dump valve opens and 3rd point heater Spill valve closes.
(2) Feedwater temperature rises

C. (1) 3rd point heater dump valve opens
3rd point heater Bleeder Trip Valve closes and Spill valve opens
(2) Feedwater temperature lowers

D (1) 2nd point heater drain valve closes
3rd point heater dump valve opens and 3rd point heater Spill valve closes
(2) Feedwater temperature lowers

Proposed Answer: C

A. Incorrect: Feedwater temperature lowers not rises. Plausible because with the loss of extraction steam to the 3rd point heater, condensate / feedwater flow going to the downstream heaters is colder. This results in more efficient extraction steam condensation in these heaters and subsequent higher extraction steam flow. However the overall effect is a drop in final feedwater temperature.

B. Incorrect: The 2nd point drain valve does not close. Plausible if the operator concludes that the system will shut off the drain from the 2nd point in an attempt to control the level in the 3rd point. However the 2nd point drain valve is controlled strictly off the level in the 2nd point. Additionally the 3rd point spill valve opens, not closes. Finally feedwater temperature lowers, not rises.

C. Correct: At 16 inches heater level the 3rd point heater dump valve opens. At 20 inches the extraction steam bleeder trip valve closes and the spill valve opens diverting extraction steam from the low pressure turbine to the main condenser.

The extraction valve going closed lowers the temperature to the downstream heaters causing an overall drop in final feedwater temperature.

D. Incorrect: The 2nd point drain valve does not close. The spill valve also does not close.

Technical Reference(s)	ARP CIC-D1 Feedwater heater reference text, page 22	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-04-09, EO 10 and 11		(As available)
Question Source:	Bank Modified Bank New	WTSI 11890	Modified for Pilgrim (Note changes or attach parent)
Question History:	Last NRC Exam:	2008	Brunswick

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 #	2	
	Topic and K/A #	202001	A4.02
	Importance Rating	3.5	

Reactor Recirculation: Ability to manually operate and/or monitor in the control room: System valves

Proposed Question: RO Question # 36

The plant is at rated conditions when Safety Valve 203-4A begins to leak. Additional information is as follows:

- Drywell pressure peaked at 4 psig and is now 3 psig and lowering.
- RPV level lowered to -10 inches before recovering and is now +30 inches.
- RPV pressure is 700 psig and lowering

Based on the above ...

Recirc Pump Discharge Valve ___ (1) ____ (MO-202-5A / MO-2025B) is closed.

In order to reopen this valve, you must first depress ___ (2) ___.

A. (1) MO-202-5A
(2) LPCI Initiation Signal Reset pushbuttons

B. (1) MO-202-5B
(2) LPCI Initiation Signal Reset pushbuttons

C. (1) MO-202-5A
(2) LPCI Loop Select Reset pushbuttons

D (1) MO-202-5B
(2) LPCI Loop Select Reset pushbuttons

Proposed Answer: D

A. Incorrect: MO-202-5B will close. MO-202-5A will remain open. Also, the LPCI Initiation Reset Pushbuttons do not need to be depressed. Plausible in that the 2.2 psig drywell pressure is also a LPCI initiation signal. However the initiation logic does not control the recirc pump discharge valves.

B. Incorrect: The LPCI Initiation Reset Pushbuttons do not need to be depressed.

C. Incorrect: MO-202-5B will close. Plausible if the operator thinks “A” loop is the default loop for injection.

D. Correct: LPCI loop select logic actuated when drywell pressure exceeded 2.2 psig. Since the initiation was based on a safety valve leak, there is no difference between the recirc riser pressures. As a result, the loop select logic will select the default loop for injection – the "B" loop. As a result, MO-202-5B was closed by the logic. In order to re-open the valve the loop select logic must be reset by depressing the loop select

Technical Reference(s)	PNPS 2.2.19, Attachment 9 RHR Ref text, page 49, page 63	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-09-01, EO-27	(As available)
Question Source:	Bank Modified Bank New X	(Note changes or attach parent)
Question History:	Last NRC Exam: N/A	

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	
	Topic and K/A #	288000	2.1.30
	Importance Rating	4.4	

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

Proposed Question: RO Question # 37

Turbine building pressure is normally maintained lower than atmospheric pressure.

Turbine building pressure can be read at manometers located ___ (1) ___ and is controlled by varying the number of Turbine Building ___ (2) ____ (Roof Exhausters / Supply Fans) in service.

A. by panel C-60
Roof Exhausters

B. by panel C--60
Supply Fans

C. inside Fan Room #1
Roof Exhausters

D inside Fan Room #1
Supply Fans

Proposed Answer: A

A. Correct: There are two manometers up by C-60. One of the two manometers next to panel C-60 reads the turbine building operating floor pressure with respect to atmosphere; the other reads condenser compartment pressure with respect to atmosphere. Turbine building roof exhaust fans (VREX-102A, B, C, D, E, F) are run as required to maintain the turbine building pressure, as read on the manometers next to panel C-60, between -0.10 and -0.20 inches H2O with the turbine building trucklock door closed.

B. Incorrect: The number of turbine building roof exhaust fans is varied to maintain the required negative turbine building pressure.

C. Incorrect: The manometers are located by panel C-60 vice Fan Room #1.

D. Incorrect: The manometers are located by panel C-60 vice Fan Room #1. Also The number of turbine building roof exhaust fans is varied to maintain the required negative turbine building pressure.

Technical Reference(s)	PNPS 2.2.39, page 9 and section 7.2	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-NL-03-08-02, EO-3d	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	Topic and K/A #	290003	A1.04
	Importance Rating	2.5	

Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROOM HVAC controls including: Control room pressure

Proposed Question: RO Question # 38

The plant is shutdown with the Control Room Ventilation system temporarily shutdown for maintenance. The maintenance has been completed and the Control Room Ventilation is about to be placed into service.

Which one of the following is correct regarding the startup of the system and the response of control room pressure?

The system is placed back in service by starting …

A. Only one supply and one exhaust fan. Control room pressure will not begin to increase until Air Mixing Box dampers are manually adjusted.

B. Two supply fans and two exhaust fans. Control room pressure will not begin to increase until Air Mixing Box dampers are manually adjusted.

C. Only one supply and one exhaust fan. Control room pressure will then increase and be positive with respect to other ventilated areas provided that manual isolation dampers are in their normal position.

D Two supply fans and two exhaust fans. Control room pressure will not increase above other areas until the 2nd set of fans is placed into service and manual isolation dampers are opened.

Proposed Answer: C

A. Incorrect: The air mixing boxes are used to control temperature, not pressure and are controlled automatically by their controllers.

B. Incorrect: Only one train of supply and exhaust fans are placed in service. The other train is in standby. Also the air mixing boxes are used to control temperature, not pressure.

C. Correct: The control room, cable spreading room, and computer room ventilation system operation is initiated by first observing temperature and damper controls are at their automatic position. Mixed air and hot and cold reset controllers are placed at their vertical position. One of two supply fans (VAC-104A or B) and recirculation/exhaust fans (VRF-101A or B) are energized with one fan in standby. The ventilated areas served are maintained at a positive pressure with respect to other station ventilation zones via isolation dampers. When the system is shutdown, these dampers are left in their normal in-service position. Therefore when placing the system back in service pressure will increase as soon as the fans are restarted.

D. Incorrect: Only one train of supply and exhaust fans are placed in service.

Technical Reference(s)	PNPS 2.2.46, section 7.1.1 CR HVAC Ref Text, page 9 for function of mixing boxes		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-NL-03-08-03, TO		(As available)
Question Source:	Bank Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41 55.43		4

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295018	AK1.01
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Effects on component/system operations

Proposed Question: RO Question # 39

Which one of the following is correct regarding the continued operation of the Recirculation System following a partial loss of the RBCCW System?

A. ONLY a loss of RBCCW LOOP “A” requires a shutdown of BOTH Recirc pumps.

B. ONLY a loss of RBCCW LOOP “B” requires a shutdown of BOTH Recirc pumps.

C. A loss of EITHER RBCCW Loop requires a shutdown of BOTH Recirc Pumps.

D A loss of RBCCW Loop “A” requires a shutdown of ONLY Recirc Pump “A”.
A loss of RBCCW Loop “B” requires a shutdown of ONLY Recirc Pump “B”.

Proposed Answer: C

A. Incorrect: Both loops are required. Plausible in that some loads are cooled by only one loop. For example, both CRD pumps are cooled by the “B” RBCCW loop.

B. Incorrect: Both loops are required. Plausible in that some loads are cooled by only one loop. For example, both CRD pumps are cooled by the “B” RBCCW loop.

C. Correct: RBCCW Loop “A” supplies both A & B Recirculation Pump MG Set Fluid Coupling Oil Coolers. Without cooling water flow the MG sets must be secured.

RBCCW Loop “B” supplies both A & B Recirculation Pump Seal Water Coolers and A & B Recirculation Motor Lube Oil Coolers. Without cooling water flow the Recirc pumps must be secured.

D. Incorrect: Each Recirc Pump and MG set requires both loops of RBCCW.

Technical Reference(s)	PNPS 2.4.42, Attachment 6	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None

Learning Objective:	O-RO-02-02-06, EO-8		(As available)
Question Source:	Bank Modified Bank New	PNPS # 12314	Editorial changes (Note changes or attach parent)
Question History:	Last NRC Exam:	Not Used	

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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and 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 # 40

EOP-03, Primary Containment Control, is being executed due to a high drywell pressure condition.

Given that Torus level is 180 inches, which one of the following would result in exceeding the Primary Containment Pressure Limit of EOP-03?

A. Drywell pressure of 35 psig

B. Drywell pressure of 56 psig

C. Torus Bottom pressure of 35 psig

D Torus Bottom pressure of 56 psig

Proposed Answer: B

A. Incorrect: The PCPL would not be exceeded at this drywell pressure and torus level. Plausible in that this would exceed the Pressure Suppression Pressure of EOP-03. The PSP is exceeded at a torus bottom pressure of ~ 32 pounds. Adding in the weight of the water, ~ 7 to 8 psig, this limit would be exceeded.

B. Correct: The PCPL is a function of two curves. One being the maximum dp across the drywell vent valves that will ensure the valves will open. This dp is 56 psig. As torus bottom pressure increases, this curve also increases since containment pressure is measured at the torus bottom. This limit increases until it reaches 60 psig. At higher torus levels, the 2nd factor becomes limiting. This limit is the overpressure rating limit of the containment while taking advantage of conservative factors built into the limit. This 2nd curve is developed by taking the design pressure of 56 pounds and increasing it by 10%. Therefore the EOP limit is 56 X 1.10 which equals 61.6 psig. This limit is rounded down to 60 psig.

With a torus level of 180 inches, the Torus Bottom Pressure would be the sum of both drywell pressure and the weight of the water in the torus. 180 inches of water is 15 feet of water. This would add ~ 7 to 8 pounds of pressure due to just the weight of water. Therefore 56 psig drywell pressure plus 7 pounds of water weight would equal 63 psig, thus exceeding the PCPL.

C. Incorrect: The limit is 60 psig torus bottom pressure. Plausible in that this would exceed the Pressure Suppression Pressure of EOP-03. The PSP is exceeded at a torus bottom pressure of ~ 32 pounds.

D. Incorrect: A torus pressure of 56 pounds would not exceed the PCPL. Plausible in that this is the design pressure of the containment.

Technical Reference(s)	PCPL graph		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-05, EO-12		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis	X	
10 CFR Part 55 Content:	55.41	9	
	55.43		

Shielding, isolation, and containment design features, including access limitations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	600000	AK1.02
	Importance Rating	2.9	

Knowledge of the operation applications of the following concepts as they apply to Plant Fire On Site: Fire Fighting

Proposed Question: RO Question # 41

The plant is at 100% when a Main Transformer Lockout results in a scram. A report of smoke issuing from the Main Transformer area is received..

Which one of the following systems should respond (1) AND if this system does not automatically respond what operator actions (2) are required?

A. (1) A Deluge Water System
(2) Manually actuated from Panel C-7

B. (1) Preaction water spray system
(2) Manually actuate locally from the Turbine Building Trucklock area

C. (1) A Deluge water system
(2) Manually actuate locally from the Turbine Building Trucklock area

D (1) Preaction water spray system
(2) Manually actuated from Panel C-7

Proposed Answer: C

A. Incorrect - Actuation is from the local Deluge station TB 23 Turbine Trucklock area

B. Incorrect - it's a Deluge system that actuates following a Main Turbine lockout when rate of rise temperature indicators trip

C. Correct - Deluge system actuates from following a Main Turbine lockout when rate of rise temperature indicators trip the Deluge valve. If the Deluge valve fails to trip, manual actuation can be accomplished locally at the Deluge station TB 23 Turbine Trucklock area

D. Incorrect - Actuation is from the local Deluge station TB 23 Turbine Trucklock area

Technical Reference(s)	PNPS 5.5.2, Att. 18, pg 83 and 86	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-10-05, TO		(As available)
Question Source:	Bank	10579	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2014	Pilgrim
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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295004	AK2.02
	Importance Rating	3.0	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF D.C. POWER and the following: Batteries

Proposed Question: RO Question # 42

A complete loss of station 125 VDC has occurred. IAW station procedure 5.3.36, Extensive Damage Mitigation Guidelines, which one of the following is correct regarding SRV operation.

Manual SRV operation ...

A. cannot be restored. Alternate depressurization methods must be utilized IAW 5.3.24, Alternate Methods for Venting and Depressurizing the RPV.

B. can be restored by connecting portable batteries to any SRV at its Alternate Shutdown Panel.

C. can be restored by connecting portable batteries to the load side of any SRV control switch at the C903 panel.

D can be restored to SRV 203B or C by connecting the backup N2 supply system in service and increasing the regulator setting until the SRV lifts.

Proposed Answer: B

A. Incorrect: PNPS 5.3.36 provides direction for hooking up portable battery carts to the SRVs at their Alternate Shutdown Panels to restore SRV capability.

B. Correct: PNPS 5.3.36 provides direction for hooking up portable battery carts to the SRVs at their Alternate Shutdown Panels to restore SRV capability.

C. Incorrect: PNPS 5.3.36 provides direction for hooking up portable battery carts to the SRVs at their Alternate Shutdown Panels to restore SRV capability.

D. Incorrect: This method is not directed. Plausible as elevated drywell pneumatic pressure can cause SRVs to open at PNPS (see PNPS 2.4.29)

Technical Reference(s)	PNPS 5.3.36, Attachment 5	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-NL-06-01-03, EO-6	(As available)

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	295001	AK2.06
	Importance Rating	3.8	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION and the following: Reactor power

Proposed Question: RO Question # 43

Plant conditions are as follows:

- 100% power
- Core Flow is 67 MLbm/hr
- The Feedwater Level Control (FWLC) system is in Single Element to support troubleshooting

Then, the FWLC total feedwater flow signal fails downscale.

After the resulting transient stabilizes, which one of the following is now required?

Note: A Power to Flow map is available for your use.

A. Insert a manual scram

B. Insert the RPR array and lower power

C. Verify Core Decay Ratios are within limits.

D Insert rods using the Reverse Order of the Pull Sheet and lower power.

Proposed Answer: B

A. Incorrect: This action is not yet specified. Plausible in that the reactor is operating in the Exclusion Zone.

B. Correct: With a Total Loss of Feedflow, the Recirc Pumps will run back to 26% speed. Given that the plant was initially on the approximately 102% load line, and using the Power To Flow Map, power will level out at approximately 53%. This will place the plant in the Exclusion Zone. Entry into 2.4.165 is then required. IAW Subsequent action #2, the control rods are to be inserted IAW Section 7.9 of 2.1.14 to clear the Exclusion Zone. Section 7.9 specifies use of the RPR array.

C. Incorrect: This would be correct if the operator concludes that the reactor is operating in the Buffer Zone. Plausible if the operator believes that the Recirc pumps will only run back to 44% speed (limiter #2) because 0.44% X 67 MLbm/hr = 29.5 MLbm/hr.

D. Entry into 2.4.165 is then required. IAW Subsequent action #2, the control rods are to be inserted IAW Section 7.9 of 2.1.14 to clear the Exclusion Zone. Section 7.9 specifies use of the RPR array.

Technical Reference(s)	2.4.165 Power to Flow Map 2.1.14, section 7.9	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: Power to Flow Map

Learning Objective: O-RO-03-03-09(02), EO-7 (As available)

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

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	5, 10
	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.

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295003	AK2.01
	Importance Rating	3.2	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF A.C. POWER and the following: Station batteries

Proposed Question: RO Question # 44

The plant is at rated conditions, with the 250 VDC system aligned for normal operation.

Then, a load reject and loss of offsite power occurs.

- Bus A6 locks out when EDG “B” attempts to re-energize the bus.
- All other systems respond as designed.
- HPCI is placed in service for pressure control
- RCIC is placed in service for level control
- No other actions have been taken

Which one of the following is correct regarding the impact on the 250 VDC system?

A. The in-service charger remained energized. The backup charger de-energized. There is no impact on battery operation as a result.

B. The in-service charger de-energized but the battery transferred to the back-up charger. There is no impact on battery operation as a result.

C. The in-service charger de-energized. Manual action must be initiated to place the backup charger in service within 8 hours to ensure continued operation of the 250 VDC battery.

D The in-service charger de-energized. Manual action must be initiated to place the backup charger in service within 12 hours to ensure continued operation of the 250 VDC battery.

Proposed Answer: C

A. Incorrect: The in-service charger de-energized. Manual action must be taken to place the back-up charger in service to prevent eventual loss of the 250 VDC system. Plausible if the operator confuses the power supplies and impact of an A6 bus lockout.

B. Incorrect: Manual action must be taken to place the back-up charger in service to prevent eventual loss of the 250 VDC system. Plausible if the operator believes that the battery will transfer to the backup charger automatically.

C. Correct: The “normal” charger, D-13, is supplied from 480 VAC MCC B-14. When A-6 locked out, B-14 was lost. The backup charger, D-15, is powered from 480 VAC MCC B-10. B-10 is powered from swing bus B-6. B-6 re-energized when EDG “A” re-energized bus A5.

The 250 VDC battery has a capacity to maintain the DC system for 8 hours at full load. With HPCI in service and no other action taken, the battery is near full capacity.

D. Incorrect: The battery is rated for 8 hours not 12 hours.

Technical Reference(s)	250 VDC Ref Text, pages 5 PNPS 2.2.13, page 10 and section 7.3.1 480 VAC Ref Text, Fig 1	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-01-01, EO-9a and b		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	295006	AK3.02
	Importance Rating	4.1	

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

Proposed Question: RO Question # 45

The reactor has scrammed following an extended run at high power conditions. Additional information is as follows:

- Rods 46-23 and 06-31 remain at position 26 and cannot be inserted
- All other control rods are inserted.
- All IRMs and SRMs are fully inserted
- Reactor power is midscale on IRM range 4 and slowly lowering
- SRM Period meters indicate -80 seconds
- Reactor pressure is 900 psig and slowly lowering

Three minutes later, Reactor Power is midscale on range 2 and continuing to lower.

Which one of the following is correct regarding the observed drop in power over the three minute period?

A. Xenon concentration is lowering and is slowing the rate of power drop.

B. The production of delayed neutrons is responsible for the rate of power drop.

C. The two partially withdrawn rods are slowing the expected rate of power drop.

D Sub-critical multiplication is maintaining the rate of power drop via source neutrons.

Proposed Answer: B

A. Incorrect: Xenon concentration is initially increasing following a scram which would add negative reactivity. Plausible if the operator is confused regarding how Xenon changes.

B. Correct: Power has dropped one decade over three minutes. Using the Poet equation this corresponds to a –80 sec period. This is the expected period following a scram due to the continued production of delayed neutrons which tend to hold up the rate of power drop.

C. Incorrect: Although the reactor may not remain shutdown under all plant conditions (Xenon free, cold conditions) is currently shutdown. This power drop is expected following a plant scram.

D. Incorrect: Delayed neutrons are retarding the rate of power drop. Plausible in that subcritical multiplication will eventually stabilize reactor power.

Technical Reference(s)	O-RO-01-02-03, slide 62, 66	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-01-02-03, Reactor Kinetics, EO-6	(As available)
Question Source:	Bank	LOR Bank
	Modified Bank	(Note changes or attach parent)
	New	
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	1
	55.43	

Fundamentals of reactor theory, including fission process, neutron multiplication, source effects, control rod effects, criticality indications, reactivity coefficients, and poison effects.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295038	EK3.02
	Importance Rating	3.9	

Knowledge of the reasons for the following responses as they apply to HIGH OFF-SITE RELEASE RATE: System isolations

Proposed Question: RO Question # 46

The plant was scrammed following rising main steam line radiation levels. Conditions have continued to degrade and current plant conditions are as follows:

- RCIC is the only injection source available
- RPV level is -100 inches and rising slowly using RCIC
- A RCIC steam leak has resulted in rising secondary containment temperatures and main stack radiation levels
- The Shift Manager declared an Alert based on the offsite release
- The RCIC Turbine Area temperature has exceeded its Maximum Safe Operating (MSO) value
- Other area temperatures are rising but they have not yet exceeded their MSO values

IAW EOP-04 and EOP-05, which one of the following is correct as to whether RCIC should be manually isolated?

A. Should be isolated because of the high offsite release

B. Should NOT be isolated because it is being used to support EOP actions

C. Should NOT be isolated because only one area has exceeded its MSO value

D Should be isolated because it is a primary system discharging and an area temperature has exceeded its MSO value

Proposed Answer: B

A. Incorrect: RCIC should NOT be isolated. Plausible in that EOP-05 entry conditions have been exceeded as evidenced by the Shift Manager declaring an Alert (the Alert levels are higher than the EOP-05 entry conditions). Exceeding the Alert levels is the trigger point for isolating discharging systems. If the operator does not think that EOP-05 has the same criteria for removing systems from service as EOP-04, the operator could conclude that the system should be isolated.

B. Correct: RCIC is being used to maintain adequate core cooling. IAW both EOP-04 and EOP-05, systems are not to be isolated if they are supporting EOP-actions.

C. Incorrect: RCIC should not be isolated because it is being used for adequate core cooling. It should remain in service regardless of the number of areas above the MSO value.

D. Incorrect: RCIC should not be isolated because it is being used for adequate core cooling.

Technical Reference(s)	EOP-05, step RR-4 EOP-04, SC-12	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-06, EO-3	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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	
	Topic and K/A #	295019	AK3.02
	Importance Rating	3.5	

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Standby air compressor operation

Proposed Question: RO Question # 47

The plant is operating at 100% power with all systems operable when the following alarm is received:

- "BACKUP DIESEL COMPRESSOR RUNNING" - C2R-D4

This alarm indicates that the air header has decreased to __ (1) ___ and that the Backup Diesel Compressor will continue to run until __ (2) ___.

A. (1) 93 psig
   (2) It is manually shutdown.

B. (1) 93 psig
   (2) It auto shuts down at 103 psig.

C. (1) 90 psig
   (2) It is manually shutdown.

D (1) 90 psig
   (2) It auto shuts down at 110 psig.

Proposed Answer: C

A. Incorrect: The compressor starts at 90 psig not 93 psig. This pressure is associated with the operation of the "lag" compressor, not the diesel air compressor.

B. Incorrect: The compressor starts at 90 psig not 93 psig. These pressures are associated with the operation of the "lag" compressor, not the diesel air compressor.

C. Correct. The backup diesel compressor starts at 90 psig and must be manually secured. When pressure reaches 110 psig the compressor unloads but remains running.

D. Incorrect: The compressor must be manually shutdown. This is the pressure at which the compressor unloads.

Technical Reference(s)	PNPS 2.2.36, section 4.5.2 and section 7.5, step [3] HPIA Reference Text page 15.	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-NL-03-09-01, EO-5	(As available)
Question Source:	Bank X	Modified 2 distractors. See comments below
	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	4
	55.43	

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295025	EA1.07
	Importance Rating	4.1	

Ability to operate and/or monitor the following as they apply to HIGH REACTOR PRESSURE: ARI/RPT/ATWS: Plant-Specific

Proposed Question: RO Question # 48

The plant was operating at 16% power when the outboard MSIVs shut. The reactor failed to scram. The SRV relief function failed and the Safety Valves are cycling on their Safety Setpoints.

Which ONE of the following is correct regarding how the Reactor Recirculation Pumps are affected by this event?

A. ONLY the Recirc MG Field Breakers trip

B. ONLY the Recirc MG Drive Motor Breakers trip

C. The Recirc MG Drive Motor Breakers AND Recirc MG Field Breakers trip

D For these conditions the Recirc MG sets remain at minimum speed.

Proposed Answer: C

A. Incorrect - The Recirc MG Drive Motor Breakers AND Recirc MG Field Breakers trip

B. Incorrect - The Recirc MG Drive Motor Breakers AND Recirc MG Field Breakers trip

C. Correct - The pressure transient caused by the MSIV closure and SRV failure will cause reactor pressure to exceed the ATWS RPV pressure setpoint of 1203 psig. The two self-actuating safety valves lift at 1280 $\pm$ 38 psig and each safety valve has a relieving capacity of 14 percent of rated steam flow.

The ATWS system will trip both the Generator Field Breaker and the MG Set Drive Motor Breaker Trips (ATWS) at an RPV pressure of1203 psig.

D. Incorrect - The Recirc MG Drive Motor Breakers AND Recirc MG Field Breakers trip

Technical Reference(s)	2.2.84, Sect. 4.3.3 and 4.3.4 Main Steam Ref Text for Safety Valve setpoints.	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-02-07-09, EO-8		(As available)
Question Source:	Bank	WTSI 11302	Updated question based on new PNPS Safety Valves
	Modified Bank		
	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	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	
	Topic and K/A #	295016	AA1.03
	Importance Rating	3.0	

Ability to operate and/or monitor the following as they apply to CONTROL ROOM ABANDONMENT : RPIS

Proposed Question: RO Question # 49

The control room has been abandoned due to a fire in the main control room. NO operator actions were taken before leaving the control room.

The reactor is then manually scrammed from outside the control room.

IAW PNPS 2.4.143, Shutdown From Outside the Control Room what is the method for determining that all control rods have inserted if the EPIC computer is NOT available?

By verifying that ....

A. the SDIVs are full utilizing the Analog Trip System instrumentation in the cable spreading room.

B. the scram air header dump valve has tripped by observing the position of the valve locally.

C. the scram inlet and out valves are both open by observing their positions at each HCU.

D all the accumulators have depressurized as indicated on local Accumulator Monitoring Panels C2222 and C2204.

Proposed Answer: C

A. Incorrect: If EPIC is not available then the backup method is to verify that all scram inlet and outlet valves are open.

B. Incorrect: If EPIC is not available then the backup method is to verify that all scram inlet and outlet valves are open.

C. Correct: If EPIC is not available then the backup method is to verify that all scram inlet and outlet valves are open.

D. Incorrect: If EPIC is not available then the backup method is to verify that all scram inlet and outlet valves are open

Technical Reference(s)	PNPS 2.4.143, subsequent actions step [16] (h) on page 12. Discussion section item [9] (f)	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RQ-03-02-07, EO-15		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	295037	EA1.09
	Importance Rating	2.8	

Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : SPDS/ERIS/CRIDS/GDS:

Proposed Question: RO Question # 50

While at rated conditions, a turbine trip occurred.

Using the attached, partial SPDS display which of the below listed actions are currently required?

Action 1: High drywell pressure initiations and isolations should be verified

Action 2: An SRV should be manually opened

Action 3: RPV water should be lowered to < -25 inches

Action 4: RPV water level should be raised and controlled between +12 and +45 inches

A. Action 4 only

B. Actions 2 and 4

C. Actions 1 and 3

D Actions 2 and 3

Proposed Answer: D

A. Incorrect: Level should not be controlled between +12 and +45 inches. It should be lowered to < -25 inches. Plausible in that the APRM DNSCL alarm tag is highlighted in red. If the operator concludes that this occurs when the APRMs are downscale than level should be controlled between +12 and +45 inches. However it actually means that power is > than the APRM downscales in the presence of a scram signal. Additionally, actual RPV pressure is 1120 psig. EOP-02 directs that pressure be stabilized below 1060 psig. Therefore pressure needs to be lowered by opening SRVs.

B. Incorrect: Level should not be controlled between +12 and +45 inches. It should be lowered to < -25 inches. Plausible in that this is the normal control band when in EOP-01.

C. Incorrect: Drywell pressure is < 2.2 psig, which is the high drywell pressure initiation setpoint. Plausible if the operator misunderstands the "Scram HI 2.2" psig alarm tag on the left hand side of the display. This alarm tag will turn red when the 2.2 psig has been exceeded. Actual drywell pressure is 1.7 psig.

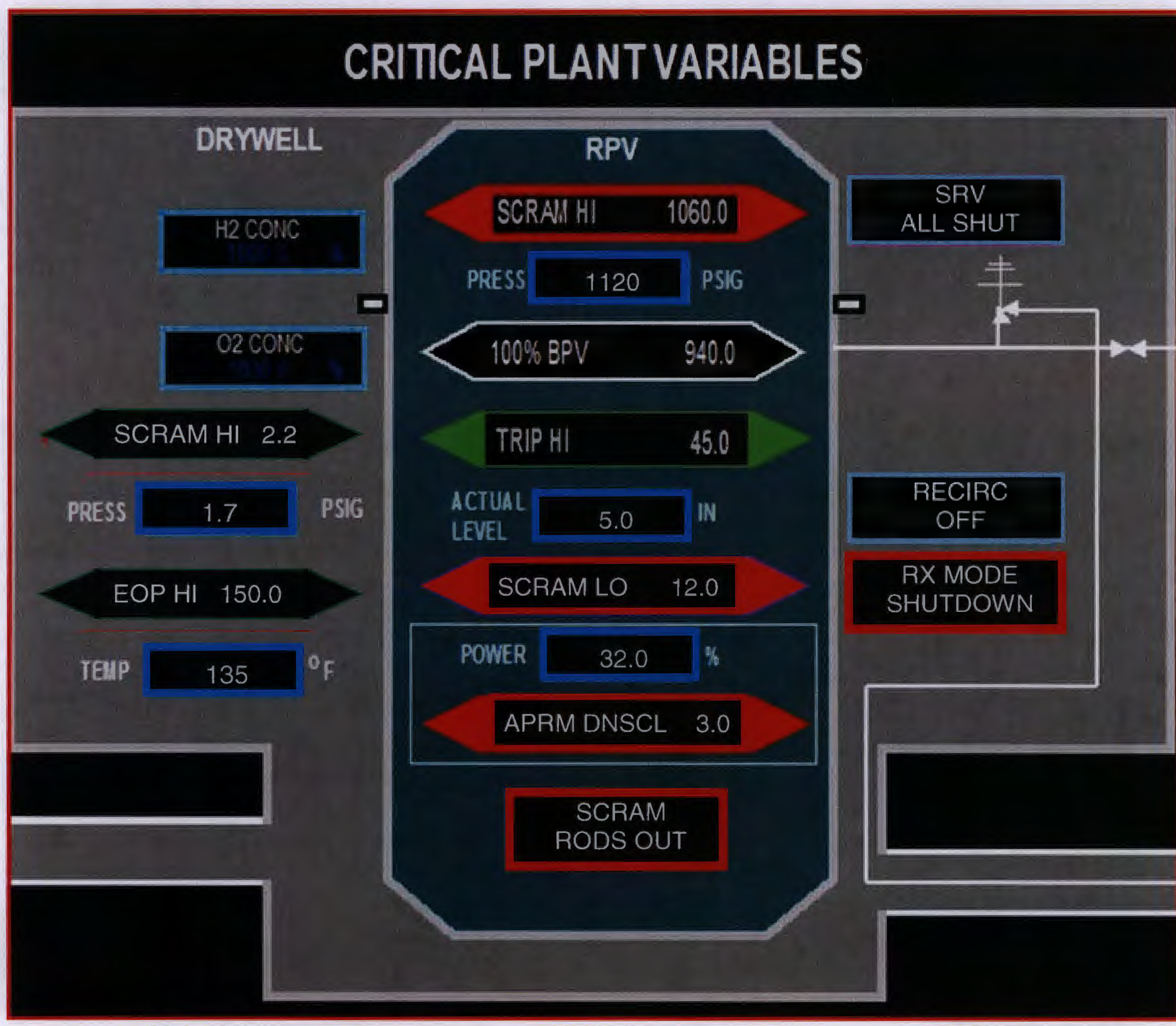
D. Correct: Actual RPV pressure is 1120 psig. EOP-02 directs that pressure be stabilized below 1060 psig. Additionally, RPV level should be lowered to < -25 inches. EOP-02 directs that when power is above 3%, and RPV level is > -25 inches then, injection is to be terminated and level lowered to -25 inches. Both of these conditions have been met in that power is 32% and RPV level is 5 inches.

Technical Reference(s)	PNPS 2.6.1 page 40 and 53 EOP-02 steps L-2 and P-5	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-04, Task 283-01-01-001	(As available)
Question Source:	Bank Modified Bank New X	(Note changes or attach parent)
Question History:	Last NRC Exam: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 55.43	10

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

Comments:

Question # 50: Partial Critical Plant Variables Display



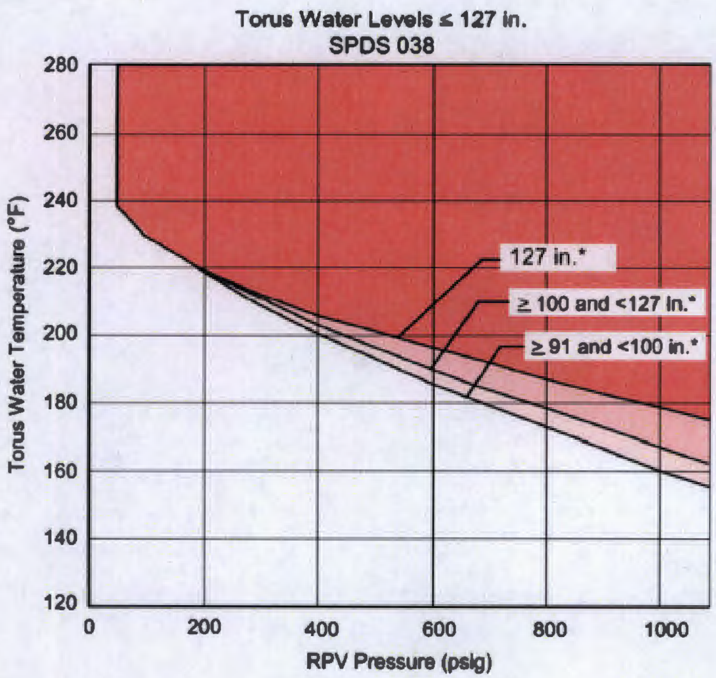
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295030	EA2.03
	Importance Rating	3.7	

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Reactor pressure

Proposed Question: RO Question # 51

A seismic event has resulted in a stuck open SRV and leak in the torus. Torus water temperature is rising steadily and RPV pressure and Torus level are both lowering steadily.

Which one of the following set of conditions would result in exceeding the limits of the Heat Capacity Temperature Limit shown below?



A. RPV pressure: 1000
Torus water level:120
Torus water Temperature: 161

B. RPV pressure: 900
Torus water level: 110
Torus water Temperature: 168

C. RPV pressure: 800
Torus water level: 95
Torus water Temperature: 177

D RPV pressure: 700
Torus water level: 110
Torus water Temperature: 181

Proposed Answer: C

A. Incorrect: The limit is not exceeded till pressure lowers to 800 and level lowers to 95 inches. Plausible in that at these values the lowest limit on the HCTL graph is being exceeded. However this limit applies for torus levels between ≥ 91 and < 100 inches.

B. Incorrect: The limit still has not been exceeded. Plausible if the operator applies the wrong limit or misreads the graph.

C. Correct: At a torus level of 95 inches, the limit shifts to the line on the graph for torus levels between ≥ 91 and < 100 inches. At 800 psig, this limit is 174 or 175 degrees. Therefore the HCTL is being exceeded and an Emergency Depressurization is now required.

D. Incorrect: At a torus level of 110 inches, the middle limit is used. At 700 pounds, the HCTL will not be exceeded until torus temperature increases to ~ 183 or 184 degrees. Plausible if the operator uses the ≥ 91 and < 100 inches limit.

Technical Reference(s)	HCTL graph		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-02, EO-22c		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	700000	AA2.04
	Importance Rating	3.6	

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: VARs outside capability curve.

Proposed Question: RO Question # 52

The plant was at full power when a partial loss of Hydrogen Seal Oil pressure occurred.

Which one of the following sets of Main Generator parameters would result in exceeding the Main Generator Capability curve?

Note: The Generator Capability Curve is provided for your use.

	Gas Pressure	MWe Loading	MVAR Loading
A.	45 psig	500 MWe	400 MVAR OUT
B.	45 psig	500 MWe	200 MVAR IN
C.	30 psig	600 MWe	100 MVAR OUT
D	30 psig	600 MWe	100 MVAR IN

Proposed Answer: B

A. Incorrect: The capability curve is not being exceeded. Plausible if the operator considers MVAR OUT as leading on the graph.

B. Correct: The MVAR limit is being exceeded. The maximum Lead Limit (IN) is ~ 100 MVAR.

C. Incorrect: At 30 psig and 600 MWe, the limit in the lagging direction is ~ 150 MVAR

D. Incorrect: At 30 psig and 600 MWe, the limit in the leading direction is 100 MVAR

Technical Reference(s)	PNPS 2.2.2, attachment 3	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		Generator Capability Curve
Learning Objective:	O-RO-03-03-03(03), EO-6	(As available)

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	295005	AA2.03
	Importance Rating	3.1	

Ability to determine and/or interpret the following as they apply to MAIN TURBINE GENERATOR TRIP : Turbine valve position

Proposed Question: RO Question # 53

The reactor was at rated conditions when a manual scram was inserted following a feedwater malfunction.

Regarding the resultant main turbine trip, which one of the following would

(1) Prevent a Turbine Lockout from occurring AND

(2) A manual action that would be required in response?

A. (1) Two Intermediate Stop Valves failing to close with all Intercept valves closing
(2) Manually transfer buses A5 and A6 to the Startup Transformer.

B. (1) Two Intermediate Stop Valves failing to close with all Intercept valves closing
(2) Manually open ACB 104 and ACB 105.

C. (1) One Intercept valve failing to close and one Intermediate Stop Valve on the same line failing to close
(2) Manually transfer buses A5 and A6 to the Startup Transformer

D (1) One Intercept valve failing to close and one Intermediate Stop Valve on the same line failing to close
(2) Manually open ACB 104 and ACB 105

Proposed Answer: D

A. Incorrect: This would not prevent a Turbine Lockout from occurring since all of the intercept valves are closed.

B. Incorrect: This would not prevent a Turbine Lockout from occurring since all of the intercept valves are closed.

C. Incorrect: A5 and A6 would transfer when the backup scram valves energized in response to the RPS trip. Plausible in that a turbine lockout will cause this transfer to occur but the turbine trip will not occur until 30 seconds after the scram. By this time the RPS would have already caused the transfer to occur.

D. Correct: If all four stop valves are closed and an IV (Intercept Valve) or ISV (Intermediate Stop Valve) is closed in ALL four steam inlet headers to the LP turbines occur, and the main disconnects are closed, a turbine lockout will result. This causes the following:

- ACB 104 & 105 to trip
- All buses transfer from UAT to SUT
- Generator field breaker trip

Since an IV and an ISV are both open in the same line, the turbine lockout will not occur. This will require ACB 104 and ACB 105 to be manually opened.

Technical Reference(s)	MHC Reference Text, section D.4 PNPS 2.2.6 section 4.3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-05-01, EO-21 and 22		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	295023	2.4.6
	Importance Rating	3.7	

Refueling Accidents: Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: RO Question # 54

The plant is refueling when a large seismic event results in a threat to the adequate cooling of the spent fuel in the Spent Fuel Pool.

IAW with EOP-04, Secondary Containment Control, and the use of Spent Fuel Pool SPRAY:

Spent Fuel Pool SPRAY is required BEFORE spent fuel pool level drops to ___ (1) ___.

The bases for initiating Spent Fuel Pool SPRAY before this level is reached is because at this level ___ (2) ___.

A. (1) 93 feet
(2) the fuel assemblies are now beginning to uncover and will begin to heatup resulting in a potential for radioactivity release and hydrogen production.

B. (1) 111 feet
(2) the fuel assemblies are now beginning to uncover and will begin to heatup resulting in a potential for radioactivity release and hydrogen production.

C. (1) 93 feet
(2) level has now lowered to 2/3's height of the fuel assemblies. Spray cooling of the fuel assemblies will be ineffective below this level.

D (1) 111 feet
(2) level has lowered to 2/3's height of the fuel assemblies. Spray cooling of the fuel assemblies will be ineffective below this level.

Proposed Answer: A

A. Correct: IAW EOP-04, , step SL-4 directs that before level lowers to this value, SFP spray be deployed

As discussed in the EOP-04 training material, irradiated fuel is a potential source of radioactivity and hydrogen whether the fuel is located in the active core or in the spent fuel pool. At a level of 93 feet, the fuel assemblies in the spent fuel pool will begin to be uncovered. As they uncover, the assemblies will begin to heatup resulting in a potential for radioactivity release and hydrogen production

B. Incorrect: The fuel assemblies will not begin to uncover until a level of 93 feet is reached. 111 feet is the point at which normal SFP makeup is augmented by the Alternate Makeup Sources of EOP-04, Table R.

C. Incorrect: At 93 feet level has just now reached the top of the fuel assemblies. Plausible if the candidate equates spray cooling of the core with spray cooling of the spent fuel pool.

D. Incorrect: The level at which spray cooling is employed is 93 feet. 111 feet is the point at which normal SFP makeup is augmented by the Alternate Makeup Sources of EOP-04, Table R.

Technical Reference(s)	EOP-04, SFP Level leg O-RO-03-04-06, Rev 10, slides 61 and 62	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	0-RQ-04-01-210, EO 9		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	295021	2.4.1
	Importance Rating	4.6	

Loss of Shutdown Cooling: Emergency Procedures / Plan: Knowledge of EOP entry conditions and immediate action steps.

Proposed Question: RO Question # 55

The plant is at 35 psig and cooling down with RHR Pump "B" in Shutdown Cooling (SDC) when a steam leak inside the drywell causes drywell pressure to rise to 3.0 psig. Operators note the following:

- Both Core Spray pumps are injecting
- RHR Pumps "A" and "C" are running on minimum flow
- RHR Pumps "B" and "D" are running and injecting via MO-1001-29B, LPCI Injection Vlv #1
- RPV water level is > +60 inches and rising rapidly on the Shutdown Level Instrument

In addition to reducing injection flow to stabilize RPV level, which one of the is correct regarding any:

(1) Additional Immediate Action that must be taken AND

(2) EOPs that should be entered?

A. (1) Manually trip RHR pumps "B" and "D" and manually isolate SDC
(2) EOP-03 ONLY

B. (1) Manually trip RHR pumps "B" and "D" and manually isolate SDC
(2) EOP-01 and EOP-03

C. (1) Secure RHR pumps "A" and "C" and re-establish a SDC lineup using one loop "B" pump
(2) EOP-03 ONLY

D (1) Secure RHR pumps "A" and "C" and re-establish a SDC lineup using one loop "B" pump
(2) EOP-01 and EOP-03

Proposed Answer: B

A. Incorrect: EOP-01 entry is also required. Plausible in that a high drywell pressure would normally be associated with EOP-03 Primary Containment Control and not necessarily RPV Control.

B. Correct: The indications are those that would occur if while RHR was in SDC mode an RHR initiation occurred with a failure of SDC to isolate. SDC should have isolated when drywell pressure rose above 2.2 psig. This should have closed both SDC suction valves and MO-1001-29A, LPCI Injection Vlv #1. Both loop "B" Pumps should have tripped when the suction valves closed. Upon failure of a group isolation to occur, immediate action is required to manually isolate the system. This would entail tripping the pumps and isolating SDC.

EOP-01 step L-1 directs the operator to verify isolations and initiations. This step requires operator to initiate the isolation if it did not occur.

A high drywell pressure of 2.2 psig requires entry into BOTH EOP-01, RPV Control, and EOP-03, Primary Containment Control. Step L-1 of EOP-01 directs that isolations be verified upon entry into EOP-01. If the isolation did not occur as expected then the operator is required to manually take the action.

C. Incorrect: SDC failed to isolate. Plausible if the operator does not recognize the SDC isolation signal. If so, the operator would want to re-establish the original SDC lineup using one RHR pump. Additionally, entry into EOP-01 would also be required.

D. Incorrect: SDC failed to isolate. Plausible if the operator does not recognize the SDC isolation signal.

Technical Reference(s)	EOP-01, EOP-03 PNPS 2.2.125.1, section 4.1, Attachment 3 O-RO-03-04-03, page 28, item "e"		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-02, EO-5		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	New	
Question History:	Last NRC Exam:	N/A	
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	
	Topic and K/A #	295026	2.4.18
	Importance Rating	3.3	

Suppression Pool High Water Temp: Emergency Procedures / Plan: Knowledge of the specific bases for EOPs.

Proposed Question: RO Question # 56

EOP-3, Primary Containment Control, directs an Emergency Depressurization if Torus temperature cannot be maintained below the Heat Capacity Temperature Limit (HCTL) curve.

Which ONE of the following identifies the basis for this action?

Performing an Emergency Depressurization at this point will ensure that the subsequent rise in ...

A. Torus level will not raise Torus Bottom pressure above the Primary Containment Pressure Limit

B. Torus temperature will not raise Torus Bottom pressure above the Pressure Suppression Pressure

C. Torus level will not raise Torus Bottom pressure above the Pressure Suppression Pressure

D Torus temperature will not raise Torus Bottom pressure above the Primary Containment Pressure Limit

Proposed Answer: D

A. Incorrect: Although torus water level will rise causing an increase in Torus Bottom pressure, it will not rise to the point necessary to exceed the HCTL.

B. Incorrect: The ED is based on preventing exceeding the PCPL.

C. Incorrect: The ED is based on preventing exceeding the PCPL.

D. Correct: As discussed in PNPS 5.3.35, the HCTL is defined as Highest Torus water temperature at which initiation of RPV depressurization will not raise the Torus bottom pressure above the Primary Containment Pressure Limit.

Technical Reference(s)	PNPS 5.3.35, section 3.0, item [19]	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-05, EO-9		(As available)
Question Source:	Bank	WTSI 16317	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Vermont Yankee
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
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	
	Topic and K/A #	295028	EK2.02
	Importance Rating	3.2	

Knowledge of the interrelations between HIGH DRYWELL TEMPERATURE and the following: Components internal to the drywell

Proposed Question: RO Question # 57

Regarding the Drywell Temperature leg of EOP-03, Primary Containment Control:

The trigger point for requiring an Emergency Depressurization is based on the .....

A. MSIV solenoid qualification temperature.

B. ADS/SRV solenoid qualification temperature.

C. primary containment design temperature.

D saturation temperature for the containment pressure that will exceed the Pressure Suppression Pressure limit.

Proposed Answer: B

A. Incorrect: The new temperature limit is based on ADS qualification.

B. Correct: The high drywell temperature emergency depressurization action level has been relaxed by eliminating consideration of the drywell design temperature. The drywell design temperature is typically based on peak conditions following the design basis primary system pipe rupture. Primary containment integrity is not expected to be immediately threatened when temperature reaches the design value; IPE and structural analyses indicate that actual drywell temperature limits are much higher, equal to or greater than the ADS qualification temperature. Replacing the emergency depressurization action level with a less restrictive value avoids unnecessary emergency depressurization, may prolong RCIC availability during extended station blackout events, and is consistent with EPG/SAG functional requirements which state that "operator actions, limits, and action levels are based on realistically bounding best-estimate engineering calculations as opposed to traditional licensing or design-basis analytical methods and assumptions"

C. Incorrect: The trigger is no longer based on the design temperature limit of 280°F.

D. Incorrect: The new temperature limit is based on ADS qualification and was not changed due to potential instrument inaccuracies.

Technical Reference(s)	O-RO-03-04-05, slide 30	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RQ-04-01-219, EO-8	(As available)
Question Source:	Bank LOR Bank Modified Bank New	(Note changes or attach parent)
Question History:	Last NRC Exam: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 55.43	

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

Comments: Based on EOP changed

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	Topic and K/A #	295031	EA2.01
	Importance Rating	4.6	

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Reactor water level

Proposed Question: RO Question # 58

Following a transient, plant conditions are:

- NO RPV injection sources are currently available
- Drywell Pressure is 8 psig
- RPV pressure is 800 psig and lowering slowly

Based on the above conditions, which one of the following is the LOWEST, stable, ACTUAL RPV Water Level (RWL) which assures Adequate Core Cooling?

A. -125" RWL.

B. -150" RWL

C. -160" RWL

D -175" RWL

Proposed Answer: C

A. Incorrect: The lowest level is -160. Plausible in that this is the top of the active fuel.

B. Incorrect: The lowest level is -160. Plausible in that this is the Minimum Steam Cooling RPV Water Level (MSCRWL)

C. Correct: With no injection, steam cooling is the core cooling mechanism. While steam cooling, the core is expected to remain adequately cooled as long as RPV water level remains above the Minimum Zero Injection Reactor Water Level (MZIRWL, -160 in.).

D. Incorrect: - 50 inches below TAF is 2/3 core height but it does NOT assure adequate core cooling unless core spray is injecting.

Technical Reference(s)	EOP-18 5.3.35 Definitions 32 and 33	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-02, EO-19, 26f		(As available)
Question Source:	Bank	LOR Bank # 70	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	N/A	
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 #	2	
	Topic and K/A #	295033	EK1.02
	Importance Rating	3.9	

Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Personnel protection

Proposed Question: RO Question # 59

The plant is at full power with fuel moves in progress on the refuel floor.

Then the following alarms and reports are received:

- REFUEL FLOOR RAD HI, C904LC-C7
- Report that the locally installed radiation monitor on the Refuel Bridge alarm is in alarm
- Report from Refuel Floor personnel of a fuel bundle drop event

IAW PNPS 5.4.3, Refueling Floor High Radiation, which one of the following actions is directed in order to protect control room personnel?

Assume all systems respond as designed.

A. Start SBGT manually

B. Start CRHEAFs manually

C. Place Control Room Ventilation in Recirculation Mode

D Don SCBAs until RP has confirmed control room rad levels

Proposed Answer: B

A. Incorrect: SBGT start will not protect control room personnel as the source of control room ventilation is from outside air.

B. Correct: Subsequent action #1 of PNPS 5.4.3 is to start CRHEAFs.

C. Incorrect: The directed action is to start CRHEAFs. Plausible due to name for this lineup. The system is normally in Recirc Mode. However this lineup does still use outside air to some extent and will not protect personnel in the control room.

D. Incorrect: The directed action is to start CRHEAFs. Plausible in that if CRHEAFs does not start then control room personnel are to don SCBAs.

Technical Reference(s)	PNPS 5.4.3, section 4.0	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO Task 273-04-01-002	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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	
	Topic and K/A #	295022	AK2.07
	Importance Rating	3.4	

Knowledge of the interrelations between LOSS OF CRD PUMPS and the following: Reactor pressure (SCRAM assist): Plant-Specific

Proposed Question: RO Question # 60

The running Control Rod Drive (CRD) pump has just tripped.

While preparing to start the standby CRD pump, an Accumulator Trouble indication is received for fully withdrawn control rod 18-41.

Before the standby pump is started an automatic reactor scram occurs.

Assuming that the accumulator for rod 18-41 has completely depressurized, which one of the following is the LOWEST reactor pressure that will ensure that normal scram times will be met for rod 18-41?

A. 1000 psig

B. 950 psig

C. 800 psig

D 450 psig

Proposed Answer: B

A. Incorrect: Normal scram times will be met at this pressure, but this is not the lowest pressure.

B. Correct: IAW PNPS 2.4.4, with the Reactor pressurized above 950 psig, normal Scram times will still be achievable regardless of accumulator pressure. The design of the CRD utilizes Reactor pressure to aid in the Scramming of control rods.

C. Incorrect: The rod will still insert at this pressure but scram times will not be met.

D. Incorrect: Scram times will not be met. This is the lowest pressure for which the rod will insert.

Technical Reference(s)	PNPS 2.4.4, section 5.0, item [5]	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-03-03-03(01), EO-7		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		6
	55.43		

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Topic and K/A #	295034	EK3.01
	Importance Rating	3.8	

Knowledge of the reasons for the following responses as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Isolating secondary containment ventilation

Proposed Question: RO Question # 61

The plant has just shutdown for refueling and refueling operations have commenced.

The Secondary Containment ventilation, high radiation isolation, is required to remain operable for the first ___ (1) ___ hours following the shutdown.

This is because ___ (2) ___.

A. (1) 24
   (2) Cold shutdown conditions are assumed to have been established within this time frame. The secondary containment high radiation exhaust isolation is not required in cold shutdown.

B. (1) 24
   (2) Exposure at the site boundary will be below the allowable TEDE limits if the fuel has decayed for at least this time without taking credit for secondary containment isolation.

C. (1) 36
   (2) Cold shutdown conditions are assumed to have been established within this time frame. The secondary containment high radiation exhaust isolation is not required in cold shutdown.

D (1) 36
   (2) Exposure at the site boundary will be below the allowable TEDE limits if the fuel has decayed for at least this time without taking credit for secondary containment isolation.

Proposed Answer: B

A. Incorrect: The secondary containment high radiation exhaust isolation is required during movement of recently irradiated fuel assemblies in the secondary containment. This is to prevent exceeding exposure limits during fuel handling accident. The requirement is not related to cold shutdown. Plausible if the candidate concludes that refueling cannot be commenced until after cold shutdown has been achieved and remembers that primary containment isolation capability is not required in cold shutdown.

B. Correct: The secondary containment ventilation high radiation isolation is required whenever secondary containment integrity is required (TS definitions). Secondary containment is required to be operable during the movement of recently irradiated fuel. Fuel assemblies that have been allowed to decay for at least 24 hours following reactor shutdown are no longer “recently irradiated.” Therefore the isolation is not required 24 hours after shutdown.

The Fuel Handling Accident (FHA) analysis is based on an alternate source term methodology (10 CFR 50.67 and R.G. 1.183). This parametric analysis concluded that the calculated TEDE values to the control room occupants, the exclusion area boundary, and the low population zone are well below the allowable TEDE limits established in 10 CFR 50.67 without crediting Secondary Containment, SGTS and CRHEAFS as long as the fuel is allowed to decay for at least 24 hours following reactor shutdown.

C. Incorrect: The isolation is required for the first 24 hours not 36 hours following the shutdown. Plausible in that 36 hours is a time frame associated with TS action statements regarding secondary containment integrity and associated systems such as SBGT and CRHEAFs.

Also, the secondary containment high radiation exhaust isolation is required during movement of recently irradiated fuel assemblies in the secondary containment. This is to prevent exceeding exposure limits. The requirement is not related to cold shutdown. Plausible if the candidate concludes that refueling cannot be commenced until after cold shutdown has been achieved and remembers that primary containment isolation capability is not required in cold shutdown.

D. Incorrect: The isolation is required for the first 24 hours not 36 hours following the shutdown. Plausible in that 36 hours is a time frame associated with TS action statements regarding secondary containment integrity and associated systems such as SBGT and CRHEAFs.

Technical Reference(s)	TS Bases page B3/4.7-20 TS Definitions page 1-5 TS Table 3.2.D		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-08-05, EO-18		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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 #	2	
	Topic and K/A #	295009	AA1.03
	Importance Rating	3.0	

Ability to operate and/or monitor the following as they apply to LOW REACTOR WATER LEVEL : Recirculation system: Plant-Specific

Proposed Question: RO Question # 62

The plant is at full power when a complete loss of normal feed occurs. RPV level lowers to -50 inches before recovering via HPCI and RCIC injection.

Which one of the following is correct regarding the status of the Recirc Pumps?

A. Recirc Pump “A”: Running at minimum speed
Recirc Pump “B”: Running at minimum speed

B. Recirc Pump “A”: Tripped
Recirc Pump “B”: Running at minimum speed

C. Recirc Pump “A”: Running at minimum speed
Recirc Pump “B”: Tripped

D Recirc Pump “A”: Tripped
Recirc Pump “B”: Tripped

Proposed Answer: D

A. Incorrect: Both pumps will trip. Plausible if the operator does not recall the ATWS trip signals and the impact of LPCI loop select. In that case the pumps would be at minimum speed due to the low feed flow signal.

B. Incorrect: Both pumps will trip. Plausible if the operator fails to consider the ATWS trip and only considers LPCI loop select which will send a trip signal to one of the pumps when activated. In this case it would have sent a trip signal to the “B” pump.

C. Incorrect: Both pumps will trip. Plausible if the operator fails to consider the ATWS trip and only considers LPCI loop select which will send a trip signal to one of the pumps when activated. In this case it would have sent a trip signal to the “B” pump.

D. Correct: Both Recirc Pumps will trip when level lowers below -46 inches due to actuation of the ATWS low level trip the Recirc Pumps.

Technical Reference(s)	PNPS 2.2.84, section 4.3.3	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-06-02, EO-18		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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 #	2	
	Topic and K/A #	295020	AA2.06
	Importance Rating	3.4	

Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : Cause of isolation

Proposed Question: RO Question # 63

The plant is at rated conditions when the following indications are received:

The inboard isolation valves for the following systems / components close while their corresponding outboard isolation valves remain open:

- RWCU
- Recirc Loop Sample
- Drywell Equipment Drain
- Drywell Floor Sump

Which one of the following is consistent with these indications?

A loss of …

A. Y-3

B. Y-4

C. RPS "A"

D RPS "B"

Proposed Answer: A

A. Correct: Y-3 supplies the isolation logic for the inboard isolation valves for these systems and components. A loss of this power supply will cause these inboard isolation valves to go close.

B. Incorrect: A loss of Y3 has occurred. Plausible in that Y4 also supplies the isolation logic for these systems /components but only for the outboard isolation valves.

C. Incorrect: A loss of Y3 has occurred. Plausible in that RPS "A" and "B" supply the power for the sensors that provide the isolation. A loss of both RPS buses would cause a complete isolation of these systems. However a loss of either one will not cause an isolation.

D. Incorrect: A loss of Y3 has occurred. Plausible in that RPS “A” and “B” supply the power for the sensors that provide the isolation. A loss of both RPS buses would cause a complete isolation of these systems. However a loss of either one will not cause an isolation.

Technical Reference(s)	PNPS 5.3.18, section 2.0		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-08-10, EO-11		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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 #	2	
	Topic and K/A #	295014	2.4.2
	Importance Rating	4.5	

Inadvertent Reactivity Addition: Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions

Proposed Question: RO Question # 64

The plant was at 5% power with the Reactor Mode Switch in Startup when the following indications and alarms are received:

- C904R-B6, MG B SPEED RATE HI alarms
- C904R-C6, MG B SPEED DEVIATION HI alarms
- All APRMs increase to 20% and are continuing to rise
- NO other annunciators are in alarm

Which one of the following is NOW required?

A. Enter EOP-01, RPV Control, and insert a manual scram.

B. Enter EOP-02, RPV Control - Failure to Scram and insert a manual scram.

C. Enter PNPS 2.4.20 Reactor Recirc System Speed or Flow Control Malfunction, lockup the Recirc MG set scoop tube and stabilize plant conditions.

D Trip Recirc Pump "B", enter PNPS 2.4.17, Recirc Pump Trip and stabilize plant conditions.

Proposed Answer: A

A. Correct: EOP-01 entry is required when a condition exists which requires a reactor scram AND following the reactor scram, reactor power remains above 3% or CANNOT be determined. With the mode switch in Startup, the APRM Hi-Hi setpoint is 15%. This entry condition has therefore been exceeded since the RPS should have scrammed the reactor.

Step R-3 of EOP-01 then requires that a manual scram be inserted.

B. Incorrect: EOP-01 is entered not EOP-02. EOP-02 is only entered if after attempting the manual scram, the resultant rod pattern cannot assure reactor shutdown under all conditions.

C. Incorrect: An RPS setpoint has been exceeded. The required action is to enter EOP-01 and scram the reactor. Plausible if the operator does not recognize the failure to scram and only addresses the speed control malfunction. PNPS 2.4.20 includes direction to lockup the affected pump's scoop tube.

D. Incorrect: An RPS setpoint has been exceeded. The required action is to enter EOP-01 and scram the reactor. Plausible if the operator does not recognize the failure to scram and only addresses the speed control malfunction. PNPS 2.4.20 includes direction to trip the pump if necessary and then enter 2.4.17

Technical Reference(s)	EOP-01, RPV Control TS Table 3.1.1 and associated notes		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-03, EO-2		(As available)
Question Source:	Bank Modified Bank New	LOR Bank	Minor changes (Note changes or attach parent)
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		10

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	Topic and K/A #	295029	2.1.32
	Importance Rating	3.8	

High Suppression Pool Water Level: Conduct of Operations: Ability to explain and apply all system limits and precautions.

Proposed Question: RO Question # 65

Drywell Sprays have been placed in service.

EOP-03 directs that Drywell Sprays be secured when torus level rises to ___ (1) ___.

This is because ___ (2) ___.

A. (1) 180 inches
(2) The reactor building to torus vacuum breakers begin to become covered at this level.

B. (1) 180 inches
(2) The torus to drywell vacuum breakers begin to become covered at this level.

C. (1) 300 inches
(2) The reactor building to torus vacuum breakers begin to become covered at this level.

D (1) 300 inches
(2) The torus to drywell vacuum breakers begin to become covered at this level.

Proposed Answer: B

A. Incorrect: The reason for securing sprays is because the torus to drywell vacuum breakers are covered at this point. Continued spray operation could result in exceeding the negative design pressure rating of the containment.

B. Correct: EOP-03 directs that drywell sprays be secured when torus level cannot be maintained below 180 inches. This is because the torus to drywell vacuum breakers are covered at this point. Continued spray operation could result in exceeding the negative design pressure rating of the containment.

C. Incorrect: The level at which drywell sprays are secured is 180 inches. Additionally the concern is based on covering the torus to drywell vacuum breakers at this point.

D. Incorrect: The level at which drywell sprays are secured is 180 inches.

Technical Reference(s)	EOP-03, steps TL-9 and 10 O-RO-03-04-05, pages 15 and 16	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-05, EO-8	(As available)

Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New	N/A	
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 #	3	
	Group #	1	
	Topic and K/A #	G1	2.1.27
	Importance Rating	3.9	

Conduct of Operations: Knowledge of system purpose and / or function.

Proposed Question: RO Question # 66

Tech Specs specify that the Rod Worth Minimizer is required ≤ 20% power but not above 20%.

This is because at higher power ...

A. fewer rod movements occur which reduces the chances of an error.

B. the Rod Block Monitor prevents fuel damage in the event of a rod drop accident.

C. adherence to the Banked Program Withdrawal Sequence prevents fuel damage in the event of a rod drop accident.

D there are no credible rod patterns that will result in a rod drop accident causing fuel enthalpy to exceed limits.

Proposed Answer: D

A. Incorrect: The RWM is not required because at high power levels there is no possible rod pattern that can result in the fuel exceeding the 280 cal/gm fuel damage limit. Plausible in that PRA analysis takes into account probabilities of an error occurring. However this process is not utilized during the analysis of a Control Rod Drop Accident (CRDA).

B. Incorrect: The RBM is designed to limit the effects of a continuous control rod withdrawal error at high power, not that of a CRDA.

C. Incorrect: Adherence to the BPWS is not required at > 20% so therefore it has no effect on limiting the effects of a rod drop accident at high power levels.

D. Correct: The purpose of the RWM is to control rod patterns during startup, such that only specified control rod sequences and relative positions are allowed over the operating range from all control rods inserted to 20% RTP. The sequences effectively limit the potential amount and rate of reactivity increase during a CRDA. When thermal power is > 20% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA.

Technical Reference(s)	TS Bases page B 3/4.3-31 and 32		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-02-07-06, EO-01		(As available)
Question Source:	Bank	WTSI 12030	Modified for PNPS and updated to PNPS Tech Specs
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2008	Cooper
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		6
	55.43		

Design, components, and function of reactivity control mechanisms and instrumentation.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	Topic and K/A #	G1	2.1.4
	Importance Rating	3.3	

Conduct of Operations: Knowledge of individual licensed operator responsibilities related to shift staffing, such as medical requirements, no-solo operation, maintenance of active license status, 10CFR55, etc.

Proposed Question: RO Question # 67

You are an active NRC Licensed Operator at PNPS.

On a scheduled day off, you went to your personal physician to discuss the results of a routine blood test.

The physician recommended that you begin taking a prescription medication in order to counter act a rising cholesterol level.

You then commence taking the medication as recommended by your physician.

Which one of the following is correct regarding your responsibility regarding the medication you are now taking? Assume that you will be taking the medication daily for at least one month.

You are ....

A. required to report the medication on your first day back at work.

B. required to report the medication but only if you take it for longer than 30 days.

C. NOT required to report the medication because cholesterol medication will not affect your fitness for duty.

D NOT required to report the medication unless you develop side effects that may affect your fitness for duty.

Proposed Answer: A

A. Correct: The operator is required to report the medication on his/her first day back at work. IAW EN-NS-102, section 5.5, employees with an active NRC Operator's License are required to report all prescription medications for evaluation in accordance with procedure EN-NS-112, 'Medical Program.'

Reporting of medications should occur on the first day of work following the medication being taken/administered.

B. Incorrect: The operator is required to report the medication on his/her first day back at work. Plausible in that 10CFR and PNPS 1.3.34 specify that the operator is required to notify the Operations Manager upon learning of a disabling illness or injury suffered by a Licensed Operator such that the NRC may be notified of the event within 30 days.

C. Incorrect: The operator is required to report the medication on his/her first day back at work.

D. Incorrect: The operator is required to report the medication on his/her first day back at work

Technical Reference(s)	EN-NS-102, section 5.5, Note on Page 48 EN-NS-112 Medical Program, section 4.11	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-06-01-09, EO-3		(As available)
Question Source:	Bank	LOR Bank # 92	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	Not used	

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 #	2	
	Topic and K/A #	G2	2.2.14
	Importance Rating	3.9	

Equipment Control: Knowledge of the process for controlling equipment configuration or status.

Proposed Question: RO Question # 68

The plant is at full power with the following conditions:

- Alarm DRYWELL/TORUS VACUUM BKR OPEN, C904LC-E1, is alarming repeatedly;
- The determination has been made that a limit switch associated with one vacuum breaker is faulty;
- The associated vacuum breaker's scanner point input to the alarm circuit is about to be disabled.

Which one of the following is correct regarding administrative actions for disabling the scanner input to the alarm as prescribed in PNPS 2.3.1, GENERAL ACTION FOR ALARM RESPONSE AND ANNUNCIATOR CONTROL?

___ (1) ___ is attached to the annunciator tile annotating that a scanner point has been disabled.

The disabled annunciator ___ (2) ___ (IS / IS NOT) required to be logged in the eSOMS eSOMS PNPS Temporary Modification Log.

A. (1) An eSOMS Caution Tag
(2) IS NOT

B. (1) A Yellow tag or sticker
(2) IS NOT

C. (1) A Yellow tag or sticker
(2) IS

D (1) An eSOMS Caution Tag
(2) IS

Proposed Answer: D

A. Incorrect: Disabling the input meets the definition of a Temp Mod and must be logged in the Temp Mod Log contained in the eSOMS Narrative Log.

B. Incorrect: A Caution Tag is used. Plausible in that a yellow sticker or tag is used if all the annunciator inputs are disabled. Additionally disabling the input meets the definition of a Temp Mod and must be logged in the Temp Mod Log contained in the eSOMS Narrative Log.

C. Incorrect: A Caution Tag is used. Plausible in that a yellow sticker or tag is used if all the annunciator inputs are disabled.

D. Correct: Because only one of the inputs is being disabled, vice the entire alarm circuit a Caution Tag is required explaining which input is disabled. Per PNPS 2.3.1 the Disabled Annunciator Control Sheet is to be filed in the Disabled Annunciator Log and because it is meets the definition of a Temporary Modification is must also be logged in the Temp Mod Log contained in the eSOMS Narrative Log.

Technical Reference(s)	PNPS 2.3.1, GENERAL ACTION FOR ALARM RESPONSE AND ANNUNCIATOR CONTROL, section 7.3 [7] (b)		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-06-06-01, EO-123		(As available)
Question Source:	Bank	PNPS Bank	Made editorial change to reduce the amount of verbiage in the distractors.
	Modified Bank		(Note changes or attach parent)
	New		
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		10
	55.43		

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	Topic and 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.

K/A Justification: Since the MHC system controls reactor pressure it is considered a reactivity control system at PNPS.

Proposed Question: RO Question # 69

Preparations are in progress for a reactor plant startup IAW PNPS 2.1.1, Startup From Shutdown. Plant conditions are:

Reactor pressure is atmospheric with the Head Vents open
Operators are aligning the MHC controls in preparation for the startup in preparation for initial rod withdrawal.

Which one of the following is correct as to how the following components are position as directed by PNPS 2.1.1?

- Speed Load Changer
- Load Limit
- MPR Setpoint
- EPR Setpoint

	Speed Load Changer	Load Limit	MPR Setpoint	EPR Setpoint
A.	0%	100%	980 psig	940 psig
B.	0%	0%	175 psig	940 psig
C.	100%	100%	175 psig	940 psig
D	0%	0%	215 psig	175 psig

Proposed Answer: B

A. Incorrect: The load limit is set to 100%. Plausible if the operator is not aware that the load limit is used to first roll the turbine off the jack. Additionally the MPR is set at 175 psig to help limit the power excursion should a reactivity event occur. Plausible in that at normal operation the MPR is normally set 40 psig above the EPR setpoint.

B. Correct: IAW 2.1.1 both the speed load changer and the load limit are positioned to accommodate the eventual turbine roll after rated pressure is achieved. The turbine is rolled first with the load limit and then brought to 1800 RPM using the speed load changer. The EPR is set to begin opening the bypass valves when full pressure of 940 psig is reached during the plant heatup. The MPR is maintained higher than reactor pressure to help limit the power and pressure excursion should a reactivity transient occur. The MPR would open the bypass valves when pressure rose to the MPR pressure setpoint.

C. Incorrect: Both the load limit and the speed load changer are both used to roll the turbine.

D. Incorrect: The MPR is set to 175 psig and the EPR at 940 psig. Plausible if the operator recalls that one of the regulators is set at 175 psig and recalls that during normal operation the other is set 40 psig higher.

Technical Reference(s)	PNPS 2.1.1, steps [21], [26], [27] and [28]	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-01-12, TO	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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 #	3	
	Topic and K/A #	G3	2.3.4
	Importance Rating	3.2	

Radiation Control: Knowledge of radiation exposure limits under normal or emergency conditions.

Proposed Question: RO Question # 70

Given the following:

- A Site Area Emergency has been declared due to a refueling accident.
- A re-entry team is being assembled to operate the bridge to move a bundle to its proper location.
- An individual assigned to the team has a lifetime accumulated dose to date of 4 Rem and an annual dose to date of 0.5 Rem.
- The Emergency Director has NOT authorized the use of the higher Emergency Exposure Limits of EP-IP-440 EMERGENCY EXPOSURE CONTROLS.

What is the MAXIMUM exposure this individual can receive without exceeding the requirements of EP-IP-440 EMERGENCY EXPOSURE CONTROLS?

A. 1.0 Rem

B. 4.5 Rem

C. 5.0 Rem

D 10.0 Rem

Proposed Answer: C

A. Incorrect: Previous exposures do not matter once the emergency is declared. Plausible in that this response is the 5 Rem minus the life time exposure

B. Incorrect: Previous exposures do not matter once the emergency is declared. Plausible in that this response is the 5 Rem minus the exposure to date

C. Correct: From EP-IP440: From the time an emergency is declared, ERO personnel are considered emergency workers. Emergency workers are allowed to receive the following exposure over the course of the emergency, exclusive of previous exposure and without special authorization: 5 Rem TEDE (whole body)

D. Incorrect: 10 Rem is the limit for operating equipment (the bridge) but is one of the higher exposure limits of EP-IP-440 that must be authorized by the ED/ EPM

Technical Reference(s)	EP-IP-440 Page 7		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-07-01-06, EO-2		(As available)
Question Source:	Bank	WTSI Bank	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2014	Pilgrim
Question Cognitive Level:	Memory or Fundamental Knowledge		
	Comprehension or Analysis		X
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 #	3	
	Topic and K/A #	G3	2.3.12
	Importance Rating	3.2	

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: RO Question # 71

An operator must enter an area with a dose rate of 1200 MR/hr to perform a task.

In accordance with EN-RP-101, Access Control For RCAs, which one of the following describes the MINIMUM monitoring and radiological controls when accessing the area?

A DLR, an Electronic Dosimeter, approved RWP and ...

(1) Continuous guarding of the entrance to prevent unauthorized entry
(2) Radiation Protection Supervision OR Lead Technician approval
(3) Continuous RP coverage

A. (2)

B. (1) and (3)

C. (2) and (3)

D (1), (2) and (3)

Proposed Answer: D

A. Incorrect - A continuous door guard and RP coverage is also required.

B. Incorrect: Radiation Protection Supervision OR Lead Technician approval is also required

C. Incorrect - A continuous door guard is also required.

D. Correct - An area that has a dose rate of 1200 mr/hr is classified as a Locked High Rad Area (LHRA). Per EN-RP-101, Section 5.5, in order to access a LHRA, each person entering a Locked High Radiation Area SHALL have a DLR, an alarming direct reading dosimeter (Electronic Dosimeter), approved RWP, RP Lead technician or RPS approval and continuous RP coverage. This procedure also specifies that while LHRAs are open, the access to the LHRA SHALL be controlled in accordance with site-specific Technical Specifications. PNPS Tech Specs specify that LHRA areas shall be locked or continuously guarded to prevent unauthorized entry

Technical Reference(s)	EN-RP-101, Section 5.5 Tech Specs Administrative Controls 5.7.2,	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-06-06-01, EO-95	(As available)

Question Source:	Bank	13722	
	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	12
	55.43	

Radiological safety principles and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	Topic and K/A #	G4	2.4.25
	Importance Rating	3.3	

Emergency Procedures / Plan: Knowledge of fire protection procedures.

Proposed Question: RO Question # 72

The on watch crew is at minimum crew complement. Which one of the following is a requirement regarding maintenance of the fire brigade?

A. All operators assigned to the fire brigade must remain within the Protected Area at all times.

B. All operators assigned to the fire brigade must remain within the Protected Area unless fire brigade response is required in the Owner Controlled Area (outside the Protected Area).

C. An operator assigned to the Fire Brigade may be dispatched into the Owner Controlled Area (outside the Protected Area) for NO LONGER than 2 hours if required provided that a way to communicate with the operator has been established.

D An operator assigned to the Fire Brigade may be dispatched into the Owner Controlled Area (outside the Protected Area) WITHOUT time restriction if required provided that a way to communicate with the operator has been established.

Proposed Answer: C

A. Incorrect – An operator maybe permitted to leave the protected area.

B. Incorrect – With minimum crew staffing only one member of the fire brigade may leave the protected area.

C. Correct – When minimum crew manning exists a fire brigade member may be dispatched into the Owner Controlled Area (outside the Protected Area) for NO LONGER than 2 hours if required provided that a way to communicate with the operator has been established.

D. Incorrect – A way to communicate with the operator must be established.

Technical Reference(s)	TS 5.2.2.d EN-OP-115, pg 70	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-06-06-01, EO 7b		(As available)
Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	Not used	
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 #	4	
	Topic and K/A #	G4	2.4.31
	Importance Rating	4.2	

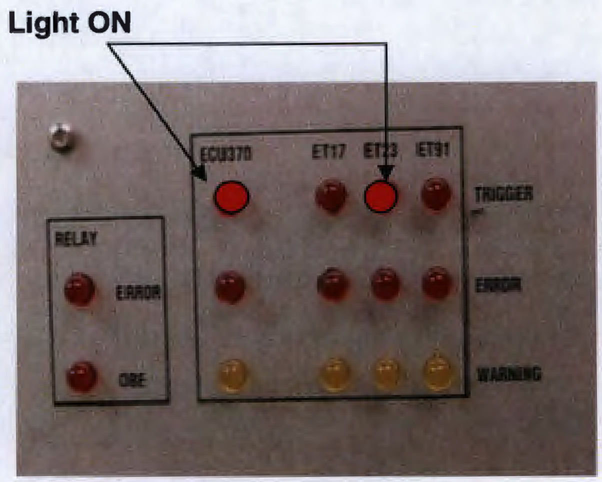
Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: RO Question # 73

Alarm "SEISMIC RECORDER OPERATING" (C903R-B1) annunciated 4 minutes ago.

The indications on the Seismic Monitoring Instrumentation are as shown in the picture to the right.

Based on these indications, one seismic monitor MUST have ....



A. exceeded the setpoint associated with the Operating Basis Earthquake (OBE).

B. exceeded the alarm setpoint but remained below the setpoint associated with the OBE.

C. failed. The yellow WARNING light would always illuminate before its associated red TRIGGER light.

D exceeded the setpoint associated with the Safe Shutdown Earthquake (SSE).

Proposed Answer: B

A. Incorrect: If the event had exceeded the OBE, the OBE light would be illuminated.

B. Correct: The middle of the 3 red lights indicates that the monitor associated with the light has exceeded its trigger setpoint (0.01g) of the alarm. The reading must be below the OBE setpoint since the OBE light is not lit.

C. Incorrect: Yellow lights are fault indicating lights. Plausible due to the nomenclature used ("warning").

D. Incorrect: The SSE setting (0.15) is higher than the OBE setting (0.08). Since the OBE light is not lit, the monitor could not have exceeded the SSE setpoint.

Technical Reference(s)	PNPS 5.2.1, section 4.0 and 5.0	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-03-24, EO-7	(As available)
Question Source:	Bank LOR # 11	
	Modified Bank	(Note changes or attach parent)
	New	
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	10
	55.43	

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

Comments:

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #	3	
	Group #	2	
	Topic and K/A #	G2	2.2.7
	Importance Rating	2.9	

Equipment Control: Knowledge of the process for conducting special or infrequent tests

Proposed Question: RO Question # 74

IAW EN-OP-116, Infrequently Performed Tests or Evolutions:

Select the choice below which correctly identifies:

(1) a test or evolution that constitutes an Infrequently Performed Test Or Evolution AND
(2) who can serve as the Senior Line Manager that has full responsibility and authority for ensuring that management's expectations are achieved during the test or evolution.

A. (1) Reactor plant startup conducted IAW PNPS 2.1.1, Startup from Shutdown following a forced outage.
(2) Any line manager senior to the Shift Manager.

B. (1) Reactor plant startup conducted IAW PNPS 2.1.1, Startup from Shutdown following a forced outage.
(2) Any qualified Shift Manager OTHER THAN the on-watch Shift Manager.

C. (1) Restoration of offsite power following a complete loss of offsite power conducted IAW PNPS 2.4.16, Distribution Alignment Electrical System Malfunctions, after the plant is stable on the EDGs and offsite power becomes available.
(2) Any line manager senior to the Shift Manager.

D (1) Restoration of offsite power following a complete loss of offsite power conducted IAW PNPS 2.4.16, Distribution Alignment Electrical System Malfunctions, after the plant is stable on the EDGs and offsite power becomes available
(2) Any qualified Shift Manager OTHER THAN the on-watch Shift Manager.

Proposed Answer: A

A. Correct: EN-OP-116, Attachment 9.1, specifically lists a Reactor Plant Startup as an IPTE. Additionally per section 3.0, item [5], the Senior Line Manager is a line manager senior to the shift manager that is designated by the General Manager Plant Operations.

B. Incorrect: The senior manager must be a line manager senior to the shift manager that is designated by the General Manager Plant Operations.

C. Incorrect: Restoration of offsite does not require an IPTE since it an evolution governed by an abnormal procedure. Plausible since it is conducted after conditions have stabilized and is an infrequently performed, complex evolution.

D. Incorrect: Restoration of offsite does not require an IPTE since it an evolution governed by an abnormal procedure. Also the senior manager must be a line manager senior to the shift manager that is designated by the General Manager Plant Operations.

Technical Reference(s)	EN-OP-116, section 3.0, item [5], Attachment 9.1, Section 5.1, item 5.1.1.2	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-06-06-01, EO-120	(As available)

Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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	
	Topic and K/A #	G1	2.1.23
	Importance Rating	4.3	

Conduct of Operations: Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: RO Question # 75

A reactor plant startup is in progress following a scram from full power two days ago. Plant conditions are as follows:

- Reactor criticality has been established
- Reactor power is midscale on IRM Range 7
- Rods are being withdrawn to establish and maintain the plant heatup rate
- Reactor pressure is 600 psig and slowly rising

Then, rod withdraw is halted in order to address an issue with the steam plant.

Five minutes later the reactor operator reports that in order to maintain the IRMs on scale, all IRMs are now midscale on range 5 and continuing to lower.

Which one of the following is now required?

A. Insert control rods to the -1% Subcritical position as listed on the Power Maneuver Plan.

B. Insert control rods to the Estimated Critical position as listed on the Power Maneuver Plan.

C. Withdraw control rods to stabilize reactor power. Re-establish criticality IAW the guidance for the initial approach to criticality.

D Withdraw control rods to stabilize reactor power. Consult Reactor Engineering for further guidance prior to withdrawing control rods.

Proposed Answer: A

A. Correct: IAW with the note on above step [14] of PNPS 2.1.4, a subcriticality event has occurred. This is because the IRMs have been down ranged two ranges and are continuing to lower. Step [14] then directs entry into attachment 1 of that procedure. Attachment 1 then directs that control rods be inserted to the -1% subcritical position as specified on the power maneuver plan.

B. Incorrect: Control rods are inserted to the -1% Subcritical position.

C. Incorrect: Control rods are to be inserted to allow for a controlled return to criticality after initial evaluation as to what has occurred.

D. Incorrect: Control rods are to be inserted to allow for a controlled return to criticality after initial evaluation as to what has occurred.

Technical Reference(s)	PNPS 2.1.4, section 7.0, step [14] and Attachment 1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-01-13, TO		(As available)
Question Source:	Bank	LOR Bank # 355	
	Modified Bank		(Note changes or attach parent)
	New		
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		10

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	Topic and K/A #	295028	EA2.02
	Importance Rating		3.9

Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE : Reactor pressure

Proposed Question: SRO Question # 76

The plant is experiencing degraded conditions. Plant conditions are as follows:

- HPCI is injecting at 4000 gpm
- HPCI is the only source of injection currently available
- The Diesel Fire Pump will be available for service in 30 minutes
- RPV pressure is 600 psig and lowering
- Actual RPV level is -130 inches and stable
- EOP-17 has just been entered when Drywell Temperature could not be maintained within the limits of EOP-03.

Which one of the following is correct regarding the RPV depressurization?

A. Remain in EOP-17 but terminate the depressurization when RPV pressure is reduced as low as practical without causing a HPCI isolation.

B. Exit EOP-17 when RPV pressure is reduced as low as practical without causing a HPCI isolation. Re-enter EOP-01 Pressure leg and stabilize RPV Pressure.

C. Remain in EOP-17 but terminate the depressurization when RPV pressure is reduced as low as practical without RPV pressure exceeding the RPV Saturation Pressure Temperature curve of EOP Caution 1.

D Exit EOP-17 when RPV pressure is reduced as low as practical without RPV pressure exceeding the RPV Saturation Pressure Temperature curve of EOP Caution 1. Re-enter EOP-01 Pressure Leg and stabilize RPV Pressure.

Proposed Answer: A

A. Correct: An Override in EOP-17 directs that if it is anticipated that the RPV depressurization will result in a loss of injection required for adequate core cooling, then
   (1) terminate RPV depressurization and
   (2) Control RPV pressure as low as practical while maintaining injection required for adequate core cooling.

   With HPCI injecting at 4000 gpm and RPV level stable at -130 inches, RPV level will quickly lower when HPCI isolates on low RPV pressure, leading to a loss of adequate core cooling.

   There is no direction to exit EOP-017. The EOP-017 override mentioned previously goes on to say that when HPCI is no longer needed for adequate core cooling then, the operator is to continue on in EOP-17 and complete the depressurization.

B. Incorrect: Once EOP-17 is entered it is only exited if RPV flooding is required or when shutdown cooling is placed in service. Pressure control would remain within EOP-17.

C. Incorrect: There is no direction to terminate the depressurization before exceeding the RPV Saturation Temperature curve. Plausible in that the rapid depress with high drywell temperature could result in RPV level indicators flashing which could cause a loss of RPV level indication. If this were to occur, EOP-17 would direct entry into EOP-16.

D. Incorrect: EOP-17 exit is not directed. Pressure control remains within EOP-17. When HPCI is no longer required for core cooling then the RPV depress is completed.

Technical Reference(s)	Correct: EOP-17, step P-1		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-09, EO 4b		
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
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
	Topic and K/A #	295018	AA2.03
	Importance Rating		3.5

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER : Cause for partial or complete loss

Proposed Question: SRO Question # 77

The plant is responding to an intake structure fouling event. Tide level is +2 feet. The following sequence then occurs:

- Shear pins for the east side travelling screens ("C" and "D" screens) shear
- Level in the east seawater pump bay begins to drop rapidly
- Level in the west seawater pump bay is normal
- Seawater Pump "A" is immediately tripped when the differential level across the "C" and "D" screens exceeds 10 feet
- The screenhouse then reports major breakthrough on "C" and "D" screens
- Level in the east seawater pump bay recovers after reaching a minimum bay level of -9 feet
- The Shift Manager directs the Control Room Supervisor to secure both Salt Service Water (SSW) Loop "B" pumps and close the rear sluice gate.

The Shift Manager directed this action because ___ (1) ___.

The most limiting Tech Spec LCO following the completion of the Shift Manager's directions is a ___ (2) ___.

A. (1) The SSW loop heat exchangers may become air bound due to the low level in the east bay.
   (2) 24 hour cold shutdown LCO

B. (1) The SSW loop heat exchangers may become air bound due to the low level in the east bay.
   (2) 72 hour LCO to restore the SSW loop

C. (1) The SSW heat exchanges may become fouled due to debris in the east bay
   (2) 24 hour cold shutdown LCO

D (1) The SSW heat exchanges may become fouled due to debris in the east bay
   (2) 72 hour LCO to restore the SSW loop

Proposed Answer: D

A. Incorrect: The design minimum level for the SSW pumps is -13'9". This level ensures that adequate NPSH is maintained. Therefore the SSW suctions were not uncovered and air could not get into the system. Additionally the most limiting LCO is a 72 hr LCO due to the SSW loop being inoperable. Plausible in that if the Ultimate Heat Sink or both SSW loops were inoperable then a 24 hr LCO would be applicable. Since level never lowered to -13'9" the UHS remained operable.

B. Incorrect: The SSW suctions were never uncovered. Additionally the most limiting LCO is a 72 hr LCO due to the SSW loop being inoperable.

C. Incorrect: The most limiting LCO is a 72 hr LCO due to the SSW loop being inoperable.

D. Correct: Differential levels across the screens of more than 8 feet can cause screen damage to occur. Screen breakthrough occurred when the differential exceeded this level. This allowed debris to enter the east seawater bay and the east SSW loop bay which serves SSW loop B. If the pumps were not secured the heat exchangers will become fouled. PNPS 2.4.154 directs that the affected SSW loop then be secured. With the loop "B" SSW pumps secured, loop "B" SSW is inoperable. This requires entry into a 72 hour LCO to restore the loop.

Technical Reference(s)	Tech Spec 3.5.B.4 PNPS 2.4.154, section 5.0 items, [1], [2] and [6] PNPS 2.4.154 subsequent action 15.	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-02-02, EO-17	(As available)

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

Question History: Last NRC Exam: N/A

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
	Topic and K/A #	700000	AA2.10
	Importance Rating		3.8

Ability to determine and/or interpret the following as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Generator overheating and the required actions.

Proposed Question: SRO Question # 78

The plant is at 100% power with the Main Generator Voltage Regulator in Automatic.

Then a disturbance on the grid causes grid voltage to slowly lower over time. The following indications are received.

- Alarm, Generator Monitor Trouble, C3R-B5, annunciates
- Several stator bar temperatures are > 65 °C and slowly rising as indicated on the Kaye Computer

Which one of the following is correct regarding:

(1) the cause of these indications AND

(2) any required action

A. (1) KVAR loading is increasing causing an increase in generator current
(2) Direct an immediate manual scram and enter PNPS 2.1.6, Reactor Scram, due to the failure of the Main Generator to trip on high temperature.

B. (1) KVAR loading is increasing causing an increase in generator current
(2) Enter PNPS 2.1.14 Station Power Changes and reduce power to lower main generator temperatures.

C. (1) MWe loading is increasing causing an increase in generator current
(2) Direct an immediate manual scram and enter PNPS 2.1.6, Reactor Scram, due to the failure of the Main Generator to trip on high temperature.

D (1) MWe loading is increasing causing an increase in generator current
(2) Enter PNPS 2.1.14 Station Power Changes and reduce power to lower main generator temperatures.

Proposed Answer: B

A. Incorrect: There is no need to insert a manual scram at this point as there is no automatic generator trip on high generator temperature. Plausible because 65 °C is the design stator bar temperature for the main generator.

B. Correct - As grid voltage lowers the voltage regulator will try to maintain its generator output voltage setpoint by raising field current. This will result in the generator picking up reactive system loads as the difference in voltage between the generator and the grid increases as grid voltage lowers.

ARP C3R-B5 requires that power be reduced if stator bar temperature exceeds 65 °C.

C. Incorrect: KVAR loading will increase. Plausible if the operator does not understand what determines the reactive loading on the main generator. Additionally, there is no need to scram at this point.

D. Incorrect: KVAR loading will increase. Plausible if the operator does not understand what determines the reactive loading on the main generator.

Technical Reference(s)	ARP C3R-B5 PNPS 2.2.2, section 7.3, Note #3.		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-03-24(01), Task 200-05-01-004		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	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 #		1
	Group #		1
	Topic and K/A #	295016	2.1.28
	Importance Rating		4.1

Control Room Abandonment: Conduct of Operations: Knowledge of the purpose and function of major system components and controls.

Proposed Question: SRO Question # 79

The plant is at rated conditions when HPCI Alternate Shutdown Panel, ASP-C155, is determined to be inoperable.

Regarding the operation of ASP-C155:

When local control is established at HPCI ASP-C-155 the control switches ___ (1) ___.

With HPCI ASP-C155 inoperable, the required Tech Spec action is to ___ (2) ___.

A. (1) install separate auto initiation circuits to ensure that fire damage will not prevent automatic ECCS system initiations.
(2) verify that fire detection and suppression is available in the cable spreading room and that fire detection is operable for those zones associated with ASP-C155.

B. (1) install separate auto initiation circuits to ensure that fire damage will prevent automatic ECCS system initiations.
(2) declare HPCI inoperable and verify that the RCIC Alternate Shutdown panels are operable

C. (1) prevent a spurious HPCI actuation that could result from signals generated from a damaged control and/or cable spreading room.
(2) verify that fire detection and suppression is available in the cable spreading room and that fire detection is operable for those zones associated with ASP-C155.

D (1) prevent a spurious HPCI actuation that could result from signals generated from a damaged control and/or cable spreading room.
(2) declare HPCI inoperable and verify that the RCIC Alternate Shutdown panels are operable

Proposed Answer: C

A. Incorrect: Separate auto initiation circuits are not installed. There are no automatic responses of these components once local control has been established.

B. Incorrect: Separate auto initiation circuits are not installed. There are no automatic responses of these components once local control has been established. Additionally it is not necessary to declare HPCI inoperable. If the fire detection or fire suppression were not operable, it still would not be necessary to declare HPCI inoperable. Instead hourly fire watches are established.

C. Correct: The switches on the ASP isolate the HPCI components from spurious signals that may occur due to fire damage. As a result there are no automatic responses of these components once local control has been established.

With the HPCI ASP inoperable the SRO must verify that fire detection and suppression is available in the cable spreading room and that fire detection is operable for those zones associated with ASP-C155.

D. Incorrect: It is not necessary to declare HPCI inoperable.

Technical Reference(s)	TS 3.12		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RQ-03-02-07, EO-1		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		2
	55.43		

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	Topic and K/A #	295001	2.1.7
	Importance Rating		4.7

Partial or Complete Loss of Forced Core Flow Circulation: Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.

Proposed Question: SRO Question # 80

Following a Recirc pump trip the following plant conditions exist:

- Reactor Power: 53%
- Core flow: 28.0 Mlbm/hr
- Reverse flow has been confirmed in the idle jet pumps

Given the above information and the attached SOLOMON print out which one of the following is correct?

Note: A Power to Flow Map is provided for your use.

A. Core Decay Ratios are NOT within limits.
IAW 2.4.165, Reactor Core Instability, raise core flow using the operating pump to exit the Buffer region.

B. Core Decay Ratios are NOT within limits.
IAW 2.4.165, Reactor Core Instability, insert control rods using Reverse Order of the Pull Sheet to exit the Buffer region.

C. Core Decay Ratios are within limits.
IAW 2.4.17, Recirc Pump Trip, maintain current plant conditions and restart the tripped Recirc pump.

D Core Decay Ratios are within limits.
IAW 2.4.165, Reactor Core Instability, raise core flow using the operating pump to exit the Buffer region.

Proposed Answer: A

A. Correct: The Core Decay Ratio is exceeding the limits of PNPS 2.4.165. This procedure requires that core flow be immediately raised (or rods inserted) to exit the Buffer Region.

B. Incorrect: If the Buffer Zone is to be exited then PNPS 2.4.165 directs that rods are to be inserted IAW 2.1.14, Section 7.9. PNPS 2.1.14, section 7.9 specifies the use of the RPR array.

C. Incorrect: The Core Decay Ratio is exceeding the limits of PNPS 2.4.165. Plausible if the operator looks at the "Previous Case" on the graph.

D. Incorrect: The Core Decay Ratio is exceeding the limits of PNPS 2.4.165. Plausible if the operator looks at the "Previous Case" on the graph. If this were true, then the second part of the distractor would be correct.

Technical Reference(s)	PNPS 2.4.17, pages 5 and 6 PNPS 2.4.165, page 5 PNPS 2.1.14, section 7.9		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Single loop power to flow map
Learning Objective:	O-RO-03-03-09, EO-7		(As available)
Question Source:	Bank Modified Bank New	LOR Bank 110	Minor Editorial changes (Note changes or attach parent)
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:

Question :

PLANT NAME; PILGRIM CYCLE XX
SOLOMON STABILITY EVALUATION REPORT

PAGE 1 OF 2
Today's Date and Time CALCULATED
Today's Date and Time PRINTED

CORE DECAY RATIO	= 0.80
HOT CHANNEL DECAY RATIO	= 0.40
CORE POWER MWT	= 1075
CORE FLOW MLB/HR	= 28.0

INITIATED BY: 3D MONICORE

CASE ID FMLD1080306064009
RESTART FRFD1080306062454

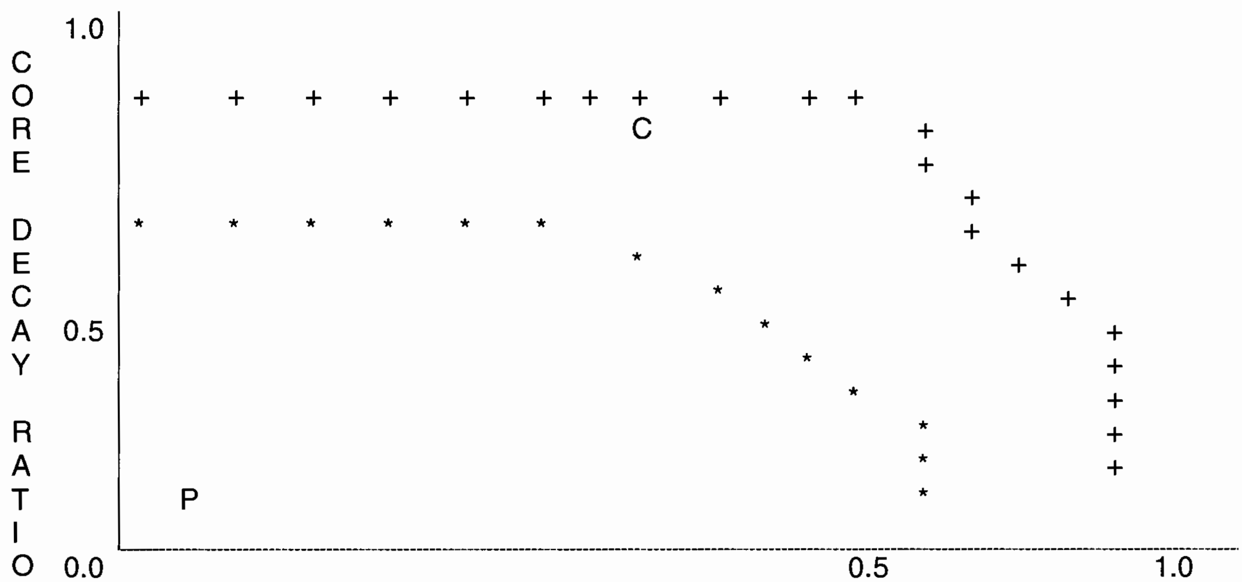
LOAD LINE SUMMARY

CORE POWER	53.0%
CORE FLOW	40.6%
LOAD LINE	94.0%

C = Current case
P = Previous

Case

STABILITY ANALYSIS TYPE: OFFICIAL



Above "+"	= Reactor Instability Expected Region
Between "+" and "*"	= Reduced Stability Margin Region
Below "*"	= Large Stability Margin Region

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	Topic and K/A #	295025	2.2.22
	Importance Rating		4.7

High Reactor Pressure: Equipment Control: Knowledge of limiting conditions for operations and safety limits.

Proposed Question: SRO Question # 81

The plant is at rated conditions when the following sequence of events occur:

- An inadvertent Group I isolation results in a high RPV pressure condition
- RPS fails to trip
- The RO places the Mode Switch to Shutdown and depresses both RPS scram pushbuttons. All control rods fail to insert
- Both Recirc pumps trip
- All SRVs and both Safety valves lift
- HPCI initiates
- All Reactor Feed Pumps trip
- Reactor pressure peaks at 1350 psig and begins to lower
- The RO depresses all four ARI Pushbuttons
- All Control rods insert
- The minimum RPV level reached was -50 inches and is now +10 inches and rising

Regarding the above sequence,

A Tech Spec Safety Limit __ (1) ___ (has / has not) been exceeded.

The highest EAL exceeded is an / a ___ (2) ___ (Alert / Site Area Emergency)

Note: An EAL chart is provided for your use.

A. has not
Alert

B. has
Alert

C. has not
Unusual Event

D has
Site Area Emergency

Proposed Answer: B

A. Incorrect: A safety limit was exceeded when RPV pressure exceeded 1340 psig. Plausible in that the ASME B&PV Code permits pressure transients up to 10% over the design pressure (110%xl250 = 1375 PSIG). If the operator recalls only this fact the operator would conclude that the safety limit was not exceeded.

B. Correct: IAW TS 2.1.4, the high reactor pressure safety limit is exceeded when steam dome pressure is > 1340 psig when irradiated fuel is in the vessel.

The highest EAL exceeded was an Alert. Two Alert EALs were exceeded. EAL SA2.1 was exceeded when an automatic scram failed to shutdown the reactor but manual action was taken to insert the control rods. The RO's initiation of ARI is considered a manual action when considering this EAL (Note 6)

HPCI initiated on a high drywell pressure signal when the safety valves lifted. This is considered a loss of the Reactor Coolant System Barrier (criterion 8 of Table F-1.) As a result Alert EAL FA1.1 is exceeded based on "any loss or any potential loss of either Fuel Clad or RCS barrier".

C. Incorrect: A safety limit was exceeded when RPV pressure exceeded 1340 psig. Additionally, an Alert EAL was exceeded. Plausible if the operator only focuses on the fission product barrier matrix. In this case the operator could conclude that only a UE is warranted based on EAL FU1.1 due to the Group I isolation failure (criterion 18). This EAL is exceeded when any loss or any potential loss of the primary containment barrier has occurred.

D. Incorrect: A Site Area Emergency EAL was not exceeded. Plausible if the operator focuses on the ATWS EALs and does not consider ARI actuation a manual action taken in response to a failure of the RPS (as opposed to the mode switch and RPS pushbuttons). In this case the operator could consider EAL SS2.1 as having been exceeded.

Technical Reference(s)	TS section 2.1 Hot EAL Chart		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			Hot EAL chart
Learning Objective:	O-RO-07-02-01, EO-2		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	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 #		1
	Group #		1
	Topic and K/A #	295005	2.2.39
	Importance Rating		4.5

Main Turbine Generator Trip: Equipment Control: Knowledge of less than or equal to one hour technical specification action statements for systems.

Proposed Question: SRO Question # 82

The plant is at rated conditions when engineering reports that a 10CFR Part 21 report has been received regarding recently installed relays. These relays are used to generate the Turbine Control Valve Fast Closure trip of the Reactor Protection System (RPS).

Engineering has concluded that three of the four RPS relays will not trip when required. Two are associated with RPS "A" and the third with RPS "B". Maintenance estimates that it will take 4 hours to replace each relay.

Which one of the following is correct regarding:

(1) RPS "A" and

(2) Whether PNPS 1.3.12, NOTIFICATION AND RECALL OF PERSONNEL, requires a 10CFR50.72 NRC notification?

Note: PNPS 1.3.12, Attachment 2 is provided for your use.

A. (1) If one of the RPS "A" relays is not replaced within 6 hours, RPS "A" must be tripped. Otherwise reduce power to < 32.5%.
(2) NO

B. (1) If one of the RPS "A" relays is not replaced within 6 hours, RPS "A" must be tripped. Otherwise reduce power to <32.5%
(2) YES

C. (1) RPS "A" must be tripped within 1 hour. Otherwise reduce power to < 32.5%.
(2) NO

D (1) RPS "A" must be tripped within 1 hour. Otherwise reduce power to < 32.5%.
(2) YES

Proposed Answer: D

A. Incorrect: Full scram capability is lost. RPS "A" must be tripped within one hour. Plausible if the operator does not realize this has occurred but recalls that if the minimum number of trip channels is not met for both trip systems, which is the case, then one trip system shall be placed in trip within 6 hours (Condition b of Note 1). However Condition C is more limiting.

Additionally, 10CFR50.72(b)(3)(v) specifies that an 8 hour report is required. See "D" below.

B. Incorrect: Full scram capability is lost. RPS "A" must be tripped within one hour.

C. Incorrect: 10CFR50.72(b)(3)(v) specifies that an 8 hour report is required.

D. Correct: RPS "A" and RPS "B" each contain two relays that are designed to trip during a Turbine fast closure event and are arranged in a one-out of-two taken twice arrangement. With both RPS "A" relays inoperable, RPS "A" will not trip during a fast closure event and a scram will not occur. TS table 3.1.1 note 1, states that there shall be two operable or tripped trip systems. Condition "C" of this note specifies that if full scram capability is lost for a given function, then RPS trip capability must be restored within one hour. Since it will take 4 hours to replace one relay this is not possible. Therefore RPS "A" must be tripped within one hour.

10CFR50.72(b)(3)(v) specifies that an 8 hour report is required for "Any event or condition that at the time of discovery could have prevented the fulfillment of the safety function of structures or systems that are needed to:
(A) Shut down the Reactor and maintain it in a safe shutdown condition."

Since full scram capability was lost, this condition has been met.

Technical Reference(s)	TS Table 3.1.1 and associated notes PNPS 1.3.12, Attachment 2	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		PNPS 1.3.12, Attachment 2
Learning Objective:	O-RO-06-06-04, EO-13	(As available)
Question Source:	Bank Modified Bank New X	(Note changes or attach parent)
Question History:	Last NRC Exam: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	

	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2, 5

Facility operating limitations in the technical specifications and their bases.

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
	Topic and K/A #	295022	AA2.02
	Importance Rating		3.4

Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS : CRD system status

Proposed Question: SRO Question # 83

The reactor is operating at rated conditions with the following conditions:

- The following control rods have been declared "slow":
    - o 14-31 was declared "slow" due to the inability to meet scram time limits
    - o 34-15 was declared "slow" due to an inoperable accumulator.
    - o Both rods are fully withdrawn
- There are no other control rod deficiencies in the core.

Then, accumulator pressure begins to drop for control rod 10-31 due to a nitrogen leak at the HCU.

(1) Which one of the following accumulator pressures would require that the accumulator be declared inoperable AND

(2) Once the accumulator is declared inoperable, the required Tech Spec Actions that would allow the plant to remain at power?

Note: References are provided for your use.

A. (1) Any pressure < 1240 psig
(2) Declare the rod slow within 1 hour. Restore the accumulator to operable status within 8 hours.

B. (1) Any pressure < 940 psig
(2) Declare the rod slow within 1 hour. Restore the accumulator to operable status within 8 hours.

C. (1) Any pressure < 1240 psig
(2) Declare the rod inoperable within 1 hour. Fully insert the control rod within the next 3 hours and disarm within 4 hours.

D (1) Any pressure < 940 psig
(2) Declare the rod inoperable within 1 hour. Fully insert the control rod within the next 3 hours and disarm within 4 hours.

Proposed Answer: D

A. Incorrect: The accumulator is inoperable when < 940 psig. Plausible in that the accumulator is maintained charged by CRD charging water. 1240 psig is the setpoint for the Low Charging Water header Pressure alarm.

Also, in order to keep the plant on line, the 10-31 must be declared inoperable and fully inserted.

B. Incorrect: In order to keep the plant on line, the 10-31 must be declared inoperable and fully inserted.

C. Incorrect: The accumulator is inoperable when < 940 psig.

D. Correct: The accumulator is inoperable when the accumulator pressure is less than 940 psig.

In order to avoid a plant shutdown, the rod cannot be declared slow because the rod is adjacent to rod 14-31. TS 3.3.C, (actions for slow control rods) requires that the plant be shutdown if two slow rods are adjacent to one another.

The only other option allowed by TS 3.3.D.B.2.2 (actions for inop accumulator), is to declare the control rod inoperable. Since there are two inop accumulators and reactor pressure is above 950 psig, this action must be accomplished within 1 hour.

TS 3.3.B.C then requires that the rod be inserted within 3 hours and disarmed within 4.

Technical Reference(s)	TS sections 3.3.B, C and D PNPS 2.2.87, section 7.6, step [8]	
Proposed Reference to be provided to applicants during examination:		TS sections 3.3.B, C and D No bases Control rod matrix
Learning Objective:	O-RO-02-06-11, EO-14	(As available)
Question Source:	Bank Modified Bank New X	(Note changes or attach parent)
Question History:	Last NRC Exam: N/A	
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 #		1
	Group #		2
	Topic and K/A #	295034	2.4.2
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of system set points, interlocks and automatic actions associated with EOP entry conditions. Secondary Containment Ventilation High Radiation

Proposed Question: SRO Question # 84

The plant is cooling down IAW 2.1.5, Controlled Shutdown from Power in preparation for a maintenance outage. Reactor pressure is 800 psig and slowly lowering.

Then the following alarms are now received:

- C904L-A6, STEAM LEAKAGE AREA TEMP HI
- C904RC-B1, RWCU INLET FLOW HI
- C904LC-B5, REACTOR BLDG VENT RAD HI
- C904LC-A5, REACTOR BLDG VENT RAD HI-HI

The BOP operator has just reported that:

- RWCU has not isolated and cannot be manually isolated
- Reactor Building ventilation remains in service

Which one of the following is correct regarding:

(1) The Reactor Building Isolation System AND

(2) Actions that are NOW required?

A. The RBIS failed to initiate. Manually initiate an RBIS.
EOP-04 and EOP-05 entry is required. Direct that CRHEAFs be initiated and all available turbine building roof exhaust fans be started.

B. The RBIS responded correctly.
Enter EOP-04 ONLY and direct that the RBIS be manually initiated.

C. The RBIS responded correctly.
Enter EOP-04 ONLY and leave Reactor Building ventilation in service.

D The RBIS failed to initiate. Manually initiate an RBIS.
EOP-04 and EOP-05 entry is required. Direct that CRHEAFs be initiated and all turbine building roof exhaust fans be secured.

Proposed Answer: B

A. Incorrect: The RBIS responded correctly. The RBIS does have a ventilation high radiation isolation signal but it is based on the radiation in the Refuel Floor Exhaust. Additionally EOP-05 entry is not required. Plausible in that there is an entry condition based on receiving the REACTOR BLDG VENT RAD HI-HI alarm but EOP-05 is not entered until the alarm has been present for 15 minutes following action taken to isolate the source. If EOP-05 was entered then these would be the required actions.

B. Correct: The RBIS responded correctly as described in "A" above. EOP-04 is entered following the receipt of the REACTOR BLDG VENT RAD HI alarm. An override in EOP-04 step SC-1 directs that if a valid REACTOR BLDG VENT RAD HI-HI alarm is also present, then the SRO is to verify or initiate a secondary containment isolation and SBGT start. Manually initiating an RBIS will cause a secondary containment isolation and start SBGT.

C. Incorrect: The SRO is to verify or initiate a secondary containment isolation and SBGT start. Manually initiating an RBIS will cause a secondary containment isolation and start SBGT. Plausible in that EOP-04 step SC-5 directs that normal reactor building ventilation be operated. Additionally an override directs that if necessary, any secondary containment isolation is to be overridden and RB ventilation be placed back in service in an effort to reduce secondary containment temperature. However this action is predicated on NOT having a leak from a primary system. RWCU constitutes a primary system.

D. Incorrect: The RBIS responded correctly as described in "A" above. Additionally EOP-05 entry is not required. Plausible in that there is an entry condition based on receiving the REACTOR BLDG VENT RAD HI-HI alarm but EOP-05 is not entered until the alarm has been present for 15 minutes following action taken to isolate the source. Finally if EOP-05 was entered then all turbine building roof exhausters would be started not secured.

Technical Reference(s)	EOP-04 EOP-05		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-03-04-06, EO-11b		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

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
	Topic and K/A #	500000	EA2.01
	Importance Rating		3.5

EA2.01 - Ability to determine and / or interpret the following as they apply to HIGH PRIMARY CONTAINMENT HYDROGEN CONCENTRATIONS: Hydrogen monitoring system availability

Proposed Question: SRO Question # 85

The plant is at 5% power in the process of shutting down IAW PNPS 2.1.5, Section F, Controlled Shutdown Without Manual Scram.

- Containment de-inerting is in progress.
- Current containment O2 concentration is 12% and rising.
- Hydrogen injection has just been secured

Then a large break LOCA occurs. Plant conditions are now:

- Torus Bottom pressure is 26 PSIG
- Drywell Temperature is 258 °F
- RPV level lowered to -100 inches before recovering.
- RPV level is now + 20 inches and stable

Which one of the following is correct regarding the status of the H2/O2 analyzers and any required action?

A. The Lower Drywell Sample point is NOT available for sampling with the current plant conditions. Place the H2/O2 analyzers in service using the remaining sample points. Direct that a PASS sample be taken to determine H2 and O2 concentration in the lower drywell IAW PNPS 5.7.4.1.12, Post-Accident Sampling System Sample from Drywell.

B. The H2/O2 analyzers cannot be placed in service with the current plant conditions and hydrogen cannot be determined to be below EOP-03 limits. Immediately commence venting and purging the containment IAW PNPS 2.2.70, Primary Containment Atmospheric Control System.

C. The Lower Drywell Sample point is NOT available for sampling with the current plant conditions. Place the H2/O2 analyzers in service using the remaining sample points and verify that hydrogen concentration is less than EOP-03 limits. A PASS sample is NOT required.

D The H2/O2 analyzers cannot be placed in service with the current plant conditions. But it can be assumed that hydrogen concentration is less than EOP-03 limits. Venting and purging of the primary containment is NOT required.

Proposed Answer: D

A. Incorrect: The H2/O2 monitors are not designed for a de-inerted environment and are not to be placed in service if this is the case. Plausible in that PNPS 2.2.133 does prohibit using the lower drywell sample point if drywell temperature is > 180°F (heat trace issue).

B. Incorrect: Because the core was not uncovered it can be assumed that H2 concentration is less than 1%. Plausible in that if concentration cannot be determined to be < 1%, then this would be the required action.

C. Incorrect: The H2/O2 monitors are not designed for a de-inerted environment and are not to be placed in service if this is the case.

D. Correct: EOP-03 directs that the H2/O2 monitors be placed in service upon entry into EOP-03. However the H2/O2 analyzers are not designed to be operated in a de-inerted environment. PNPS 2.2.133 directs that if the H2O2 Monitoring System is unavailable due to de-inerted Drywell conditions, then sample Drywell and Torus for H2 and O2 in accordance with PNPS 5.7.3.2. EOP-03 provides similar directions if the monitors are not available.

This Procedure provides guidance to Operations for sampling the Drywell and Torus atmospheres for combustible gas (oxygen and hydrogen) when the hydrogen/oxygen analyzers are not available. The procedure provides the following guidance:

Failure or unavailability of the hydrogen or oxygen monitoring system does not necessarily mean that hydrogen and oxygen concentrations cannot be determined. Rather, Operator judgment is required when making that decision after examining related plant conditions. If the Operator has reasonable assurance that a transient has not occurred which has uncovered the core (i.e., water level has not dropped below TAF), then it is likely that significant hydrogen has not been produced.

IAW PNPS 5.7.3.2, Primary Containment hydrogen cannot be determined to be below 1% (EOP-03 limit) when:

- Hydrogen concentration cannot be determined within the next hour and any of the following conditions have occurred or currently exist: (Note: A pass sample is not allowed until one hour after the event)
  - Initial H2 concentration was greater than 1% prior to the current event/transient. (Not the case)
  - RPV water level has dropped below top of active fuel (Not the case).
  - RPV water level cannot be determined (Not the case).
  - System leaks with a potential for buildup of H2 within the Primary Containment from hydrogen injection (Not the case since H2 injection was isolated before the event).

Therefore H2 concentration can be assumed to be below 1% and venting is not required. Venting would be required if H2 concentration was 1%.

Technical Reference(s)	EOP-03, steps G-1 through G-4 PNPS 2.2.133 section 7.3, Cautions PNPS 5.7.3.2, section 3.0, discussion.	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-05, EO-14a and b	(As available)
Question Source:	Bank Modified Bank New X	 (Note changes or attach parent)
Question History:	Last NRC Exam: N/A	
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
	Topic and K/A #	203000	A2.16
	Importance Rating		4.5

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: Loss of coolant accident

Proposed Question: SRO Question # 86

A small leak in the drywell has occurred concurrent with a loss of offsite power. Plant conditions are as follows:

- RPV level is being controlled in its normal band via HPCI injection
- RPV pressure is 600 psig and lowering
- Drywell and Torus sprays have been initiated utilizing RHR pumps “A” and “B”.
- RHR pumps “C” and “D” started but have been shutdown. Both Core Spray pumps failed to start and are not available.
- Torus bottom pressure is currently 12 psig and lowering

Then, the leak inside the drywell increases significantly. Three minutes later plant conditions are:

- Actual RPV level is -160 inches and lowering slowly
- RPV pressure 50 psig and lowering slowly
- Torus bottom pressure 45 psig and rising slowly

Assuming that NO operator action has occurred, which one of the following is correct regarding the current status of RHR and the required action?

A. All four RHR pumps are running and injecting. Drywell and torus sprays have isolated. Re-establish drywell and torus sprays while maintaining RHR injection flow > 3600 gpm.

B. All four RHR pumps are running in injection, drywell spray and torus spray modes. Secure drywell and torus sprays and maximize RPV injection.

C. Only RHR pumps “A” and “B” are injecting. Drywell and torus sprays have isolated. Start RHR pumps “C” and “D”. Re-establish drywell and torus sprays while maintaining RHR injection flow > 3600 gpm.

D Only RHR pumps “A” and “B” are injecting. Drywell and torus sprays have isolated. Start RHR pumps “C” and “D” and maximize RPV injection.

Proposed Answer: D

A. Incorrect: Only RHR pumps "A" and "B" are running. Once the pumps have been shutdown in the presence of an initiation signal they will remain shutdown until manually started. Plausible if the operator concludes that pressure dropping below 400 psig with a low level condition will restart the pumps.

Additionally with level below -150 inches and lowering, only Core Spray can be used for spray cooling. Plausible in that the flow limit Core Spray when being used for spray cooling is 3600 gpm.

B. Incorrect: Only RHR pumps "A" and "B" are running.

C. Incorrect: Only Core Spray can be used for spray cooling. Plausible in that the Core Spray flow limit when being used for spray cooling is 3600 gpm.

D. Correct: Only RHR pumps "A" and "B" are running. Once the pumps have been shutdown in the presence of an initiation signal they will remain shutdown until manually started. Drywell and torus sprays isolated when RPV level lowered below -150 inches.

With RPV level below -150 inches and continuing to lower, and with core spray not available, the required action per EOP-01, step L-23 is to maximize injection with all sources. This will require starting RHR pumps "C" and "D" for injection.

Technical Reference(s)	PNPS 2.2.19, section 4.2.5, item [1[EOP-01, steps L-18 through L-23	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-03, EOP-01, EO-18	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41 55.43	

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
	Topic and K/A #	217000	A2.01
	Importance Rating		3.7

Ability to (a) predict the impacts of the following on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: System initiation signal

Proposed Question: SRO Question # 87

The plant is at rated conditions when I&C reports that Analog Trip System Level Indicating Switch, LIS-263-72A, will NOT trip regardless of the magnitude of the test signal utilized. I&C further reports that:

- LIS-263-72A provides an input into the RCIC low level initiation logic
- All other RPV level trip units will trip within procedure limits

Considering ONLY the impact on the RCIC System,

(1) Will RCIC still initiate on a valid low RPV low level condition AND

(2) IAW Tech Specs, which one of the following statements is correct regarding continued plant operation?

A. (1) Yes
(2) The reactor MUST be placed in cold shutdown within 24 hours if the RPV level switch is NOT repaired beforehand

B. (1) No
(2) The reactor MUST be placed in cold shutdown within 24 hours if the RPV level switch is NOT repaired beforehand

C. (1) Yes
(2) RCIC CAN be declared inoperable and continued reactor operation is permitted for the next 14 days

D (1) No
(2) RCIC CAN be declared inoperable and continued reactor operation is permitted for the next 14 days

Proposed Answer: C

A. Incorrect: RCIC can be declared inoperable which would result in a 14 day LCO by TS 3.5.D.

B. Incorrect: RCIC will still initiate. There are four level switches that input into the RCIC initiation logic. The RCIC system is initiated upon receipt of a reactor vessel low-low water level signal utilizing a one-out-of-two taken twice logic. Therefore one level switch that failed to trip would not prevent the system from initiating. Plausible in that some functions are a two out two taken once (ex: RCIC High Level shutdown).

Additionally, RCIC can be declared inoperable which would result in a 14 day LCO by 3.5.D.

C. Correct: RCIC would still initiate. There are four level switches that input into the RCIC initiation logic. Therefore one level switch that failed to trip would not prevent the system from initiating.

Note 1 to table 3.2.B, states that whenever any CSCS subsystem is required by Section 3.5 to be operable, there shall be two operable trip systems. If the first column cannot be met for one of the trip systems, that system shall be repaired or the reactor shall be placed in the Cold Shutdown Condition within 24 hours after this trip system is made or found to be inoperable. With one switch inoperable this requirement CANNOT be met. However RCIC can be declared inoperable which would result in a 14 day LCO by 3.5.D and then the 24 hour cold shutdown requirement of Note 1 would not apply.

D. Incorrect: RCIC will still initiate.

Technical Reference(s)	Tech Spec TABLE 3.2.B INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS Tech Spec 3.5.D RCIC Reference Text page 8	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-02-09-04, EO-17	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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
	Topic and K/A #	300000	2.4.46
	Importance Rating		4.2

Emergency Procedures / Plan: Ability to verify that the alarms are consistent with the plant conditions. Instrument Air

Proposed Question: SRO Question # 88

The plant is at full power with all systems aligned in their normal full power configuration.

Then, an instrument air leak causes instrument air header pressure to lower to 80 psig.

(1) Which of the alarms below would be consistent with these conditions

- AIR/N2 TO DRYWELL TROUBLE, C904LC-F3
- SPVAH PRESSURE LO, C905R-F1
- NONESSENTIAL INSTRUMENT AIR HEADER ISOLATED, C2R-B5

AND

(2) What would be a required action for that alarm?

A. (1) SPVAH PRESSURE LO
(2) Enter PNPS 5.3.8, Loss of Instrument Air, insert a manual scram and perform PNPS 2.1.6, Reactor Scram concurrently.

B. (1) AIR/N2 TO DRYWELL TROUBLE
(2) Enter PNPS 2.4.21, Double-Ended Break of the 3-inch Instrument Pneumatic Line in the Drywell, and close AO-4356, N2/Air Isolation Valve to the drywell.

C. (1) AIR/N2 TO DRYWELL TROUBLE
(2) Enter PNPS 2.2.105, Backup Nitrogen Supply System, and lineup the Backup Nitrogen Supply to the drywell.

D (1) NONESSENTIAL INSTRUMENT AIR HEADER ISOLATED
(2) Enter PNPS 5.3.8, Loss of Instrument Air, and SHUT DOWN ETS using the SHUTDOWN switch on Panel CP600 or ETS Control Panel C613.

Proposed Answer: D

A. Incorrect: The SPVAH low pressure alarm comes in at 65 psig. And would not annunciate for an 80 psig instrument air pressure.

B. Incorrect: This alarm would not be valid for the conditions presented. Plausible in that the alarm will annunciate when the pneumatic supply to the drywell lowers to 102 psig. However during normal operation, Nitrogen not air is aligned to the drywell. Therefore this alarm would annunciate on low nitrogen pressure not low air pressure.

C. Incorrect: This alarm would not be valid for the conditions presented. Plausible in that the alarm will annunciate when the pneumatic supply to the drywell lowers to 102 psig. However during normal operation, Nitrogen not air is aligned to the drywell. Therefore this alarm would annunciate on low nitrogen pressure not low air pressure.

D. Correct: When Instrument air pressure lowers to 80 psig, AO-4365, Non-Ess Instr Air Hdr Blk Vlv, closes to isolate the nonessential Instrument Air header. This will result in this alarm. The lowering air pressure would also result in entry into PNPS 5.3.38. Subsequent action step 14, directs that if AO-4365 is closed, then ETS is to be shutdown using the SHUTDOWN switch on Panel CP600 or ETS Control Panel C613.

Technical Reference(s)	PNPS 5.3.8, Section 2.0, Subsequent actions, step 1, 14 and Attachment 1	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-NL-03-09-01, EO-18	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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
	Topic and K/A #	262001	2.4.11
	Importance Rating		4.2

Emergency Procedures / Plan: Knowledge of abnormal condition procedures. AC Electrical Distribution

Proposed Question: SRO Question # 89

Pilgrim is at 100% power with all buses powered by the Unit Aux Transformer when ISO New England notifies the Control Room that the 345kV feed to the plant cannot be maintained ≥ 343.5kV if Pilgrim were to scram. Which one of the following is required?

A. Declare the Startup Transformer inoperable until informed that 343.5kv can be assured following a scram.

B. Declare the Startup Transformer inoperable and start and load both diesel generators. The Startup Transformer LCO can be exited once A5 and A6 are on the diesels.

C. Declare the A5 and A6 degraded voltage schemes inoperable until informed that 343.5kv can be assured following a scram.

D Enter a tracking LCO for the Startup Transformer. If system voltage lowers to 343.5KV, declare the Startup Transformer inoperable.

Proposed Answer: A

A. Correct – PNPS 2.4.144, Attachment 1 requires that IF ISO New England/Eversource notifies the Control Room that the 345kV feed to the plant cannot be maintained ≥ 343.5kV following a postulated Turbine trip (of PNPS) OR PNPS post-trip voltage stability cannot be predicted, THEN □ DECLARE the Startup Transformer inoperable

B. Incorrect - INOP until informed ISO can maintain 343.5KV post scram

C. Incorrect - only true when initially on the SUT.
D. Incorrect – active LCO is required.

Technical Reference(s)	PNPS 2.4.144	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-03-04(01),EO-9	(As available)

Question Source:	Bank	X	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	Not used	

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

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
	Topic and K/A #	400000	2.2.37
	Importance Rating		4.6

Component Cooling Water: Ability to determine operability and/or availability of safety related equipment.

Proposed Question: SRO Question # 90

The plant is at rated conditions. A Condition Report has just been generated documenting that the total RBCCW Loop "A" leakage has now risen from 230 to 250 dpm.

Assuming that no corrective actions have yet been implemented, this Condition Report should be coded as RBCCW Loop "A" being ....

A. Inoperable

B. Operable DNC

C. Operable – Op Eval

D Equipment Non-Functional

Proposed Answer: A

A. Correct: In order to meet the required 30-day mission time with no credit taken for non-safety related head tank makeup, total RBCCW leakage must be less than 240 dpm for each loop.

EN-OP-104 specifically directs that the Immediate Determination of Operability should consider mission time. Since the 30 day mission time cannot be met the loop is inoperable.

B. Incorrect: The code should be Inoperable. Plausible if that the operator considers the system merely degraded and still capable of performing its safety function.

C. Incorrect: The code should be Inoperable. Plausible if the operator believes that additional engineering input is required to enhance the initial operability determination. However the system is inoperable based on mission time.

D. Incorrect: The code should be Inoperable. Plausible if the operator recalls that functionality reviews are performed for non Tech Spec components and misapplies this to system requirements not addressed specifically in Tech Specs.

Technical Reference(s)	PNPS 2.2.30, section 5.3, Precaution [6] EN-OP-104, section 5.3		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RQ-04-01-230, EO-2		(As available)
Question Source:	Bank	LOR Bank	Modified stem for closed reference
	Modified Bank		(Note changes or attach parent)
	New		
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
	Topic and K/A #	234000	A2.02
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of refueling platform air system

Proposed Question: SRO Question # 91

A core off-load is in progress during a refueling outage. A fuel assembly is loaded onto the grapple when the two service air lines supplying the fuel grapple are accidently severed.

(1) How does this affect the operation of the fuel grapple AND

(2) What action is required to safely manage and control any repair efforts?

A. (1) The grapple can still be engaged BUT cannot be disengaged
(2) If the bridge is over the core, the bridge must be moved to the spent fuel pool IAW PNPS 4.3, Fuel Handling, prior to attempting repairs.

B. (1) The grapple cannot be engaged or disengaged.
(2) If the bridge is over the core, the bridge must be moved to another location IAW PNPS 2.2.75, prior to attempting repairs.

C. (1) The grapple can still be engaged BUT cannot be disengaged
(2) Provided that the Shift Manager has granted permission, repair efforts can begin with the bridge over the core if the requirements of EN-MA-118, Foreign Material Exclusion, are met for any tools or materials utilized.

D (1) The grapple cannot be engaged or disengaged.
(2) Provided that the Shift Manager has granted permission, repair efforts can begin with the bridge over the core if the requirements of EN-MA-118, Foreign Material Exclusion, are met for any tools or materials utilized.

Proposed Answer: B

A. Incorrect: The grapple cannot be engaged either.

B. Correct: The grapple hooks fail closed as-is on loss of air supply. If the grapple is disengaged at the time of the air loss it cannot be engaged. If disengaged it cannot be engaged. PNPS 4.3, specifies that repairs cannot be attempted with the refuel bridge over the core unless the failure prevented movement of the bridge. A loss of air will not prevent movement of the bridge.

C. Incorrect: The grapple cannot be engaged either. Additionally PNPS 4.3, prohibits conducting repairs with the bridge over the core.

D. Incorrect: PNPS 4.3, prohibits conducting repairs with the bridge over the core.

Technical Reference(s)	PNPS 4.3, precaution [9] FSAR page 7.6.2		(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-06-04-03, EO-9g		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41		
	55.43		7

Fuel handling facilities and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	Topic and K/A #	202001	2.4.31
	Importance Rating		4.1

Recirculation: Knowledge of annunciator alarms, indications, or response procedures.

Proposed Question: SRO Question # 92

PNPS is at 50% power with the "A" Recirculation Pump secured when the following alarms are received:

- C3LC-A7, B-2 TRIP
- C3LC-A8, B-2 OVERLOAD

Which one of the following is required?

A. Enter 2.1.6 and direct a manual scram
Cross-tie RBCCW loops
Declare a 24 hour Cold Shutdown LCO

B. Enter 2.1.6 and direct a manual scram
Cross-tie RBCCW loops
Declare a 7 day Cold Shutdown LCO

C. Commence a Reactor Shutdown per 2.1.5, Controlled Shutdown From Power
Cross-tie RBCCW loops
Declare a 24 hour Cold Shutdown LCO

D Commence a Reactor Shutdown per 2.1.5, Controlled Shutdown From Power
Cross-tie RBCCW loops
Declare a 7 day Cold Shutdown LCO

Proposed Answer: A

A. Correct: Recirc "B" will trip on the loss of B2 following a loss of both AC oil pumps.. This will place the reactor on natural circulation. Procedure 2.4.17 will then direct a reactor scram. Procedure 2.4.B2 will direct that RBCCW be cross-tied. This will require entry into a 24 hour LCO.

B. Incorrect: A 24 hour LCO is required.

C. Incorrect: A reactor scram is required.

D. Incorrect: A reactor scram is required. Also a 24 hour LCO is required.

Technical Reference(s)	2.4.B2 2.4.17 Bases for Tech Spec 3.5.B.3		(Attach if not previously provided)
Proposed Reference 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:	Last NRC Exam:	2009	Pilgrim

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
	Topic and K/A #	214000	2.1.20
	Importance Rating		4.6

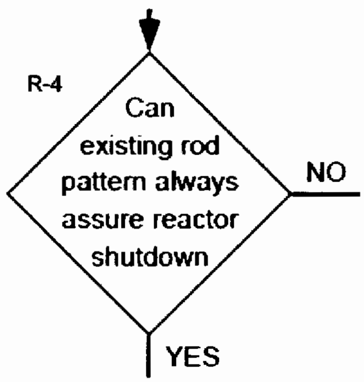
Conduct of Operations: Ability to interpret and execute procedure steps. RPIS

Proposed Question: SRO Question # 93

EOP-01, has just been entered when drywell pressure rose to 3.5 psig. The At-The-Controls operator reports the following:

- Two rods are at position 04
- Another rod indicates "Black-Black", and its associated Full-In light is NOT illuminated
- The Call Rods display is magenta
- Reactor Power is on Range 4 of the fully inserted IRMs and lowering
- The MSIVs are open
- RPV level is +15 inches and rising slowly
- RPV pressure is stable at 945 psig

Regarding the following EOP-01 decision element and the information provided,



(1) What is the correct response to step R-4 above AND

(2) What actions are now required regarding RPV pressure control?

A. (1) YES
(2) Continue on in EOP-01 and direct that one turbine bypass valve be open fully and RPV pressure lowered and stabilized in a 450 – 550 psig band.

B. (1) YES
(2) Continue on in EOP-01 and direct that one SRV be opened and RPV pressure lowered and stabilized in a 450 – 550 psig band.

C. (1) NO
(2) Exit EOP-01 and enter EOP-02. Inhibit ADS and bypass the MSIV low RPV water level isolation. Direct a controlled cooldown be commenced using the turbine bypass valves.

D (1) NO
(2) Exit EOP-01 and enter EOP-02. Do NOT Inhibit ADS or bypass the MSIV low RPV water level isolation. Direct a controlled cooldown be commenced using the turbine bypass valves.

Proposed Answer: C

A. Incorrect: The reactor may not remain shutdown under all conditions with the existing pattern. Plausible if the operator thinks that one rod can be fully withdrawn with all other rods at position 04 (the max subcritical bank withdrawal position). Additionally if pressure control were to remain in EOP-01, this would be the required pressure control strategy as specified in procedure 5.3.35.2.

B. Incorrect: The reactor may not remain shutdown under all conditions with the existing pattern.

C. Correct: The reactor may not remain shutdown under all conditions with the existing pattern. Procedure 5.3.5 defines “shutdown under all conditions” as being met if one of the following is satisfied:

- All control rods are inserted to position "04" or beyond.
- All control rods at position "00" except one control rod may be full out (position 48).
- Cold Shutdown boron weight has been injected into the RPV.
- When determined by Reactor Engineering analysis

With the position of one rod unknown and Call Rods unavailable (as indicated by the display being colored magenta) it must be assumed that the rod is not inserted to at least position 04. Therefore the answer to the decision element must be NO. EOP-01 then directs that EOP-02 be entered. EOP-02 directs ADS be inhibited and that if any MSIV is open, then the low level isolation is to be bypassed.

Steps P-6 and P-8 direct that if the reactor is currently shutdown then a cooldown is to be commenced. As applied to the Reactor, the reactor is currently shutdown if it is subcritical with Reactor power below the heating range (on scale on IRM range 7 or below).

D. Incorrect: ADS is inhibited and the MSIV isolation is bypassed.

Technical Reference(s)	PNPS 5.3.35, definitions [41] and [42] EOP-02 steps L-3, L-4, P-6 and P-8 EOP-01, step R-4	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-04, EO-16, 24	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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
	Topic and K/A #	G1	2.1.5
	Importance Rating		3.9

Conduct of Operations: Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.

Proposed Question: SRO Question # 94

The plant is operating at power.

Which of the following situations would justify the shift manager initiating a Work Hour Limits Waiver IAW EN-OM-123, Fatigue Management Program?

Note: Assume that each of these situations would result in the operator(s) exceeding the 10CFR26 Work Hour Limits.

Situation 1 Holding shift operators over to assist in restoring the plant to rated conditions after the plant was placed in Single Loop operations four days ago.

Situation 2: Holding an operator over past the operator's end of shift to complete a Tech Spec scheduled surveillance so as to avoid a turnover in the middle of the surveillance.

Situation 3: Holding shift operators over to assist in restoring the plant equipment required to exit a 12 hour hot shutdown LCO.

A. Situation 1

B. Situation 2

C. Situation 3

D Both situations 1 AND 3

Proposed Answer: C

A. Incorrect: This condition would not justify a waiver. The requirements for single loop must have been completed since those must be completed within 24 hours. Therefore restoring the plant would not constitute mitigating a Tech Spec action.

B. Incorrect: Situation 2 does not justify a waiver. This condition could have been reasonably forecasted and also does not meet any of the examples described as a condition adverse to safety.

C. Correct: IAW EN-OM-123 waivers are only permitted in situations involving a Condition Adverse to Safety or Security (3.0 [8]) AND should only be granted to address circumstances that could not have been reasonably controlled. Condition Adverse to Safety are defined as:

A situation which may be eligible for a licensee-approved waiver of work hour controls to prevent or mitigate the condition. Examples of Conditions Adverse to Safety include:

1. Public or station personnel health or safety is jeopardized.
2. Recovery from a challenge to the safety function of a system or component is delayed.
3. Compliance with another NRC regulatory requirement is impaired or prevented.
4. Mitigation of a Technical Specification required reactor shutdown or power reduction is impaired or prevented.
5. Unplanned increase in the plant status risk color assignment.
6. Compliance with site environmental permits is impaired or prevented.
7. External events (weather, fire, flooding) pose a risk to station personnel.

Situation 3 meets example #4. Neither of the other two meet any of the other examples.

D. Incorrect: Situation 1 condition would not justify a waiver.

Technical Reference(s)	EN-OM-123, definition [8] and Section 5.9		Editorial changes only
Proposed Reference to be provided to applicants during examination:			None
Learning Objective:	O-RO-06-06-01, EO-29a		(As available)
Question Source:	Bank	WTSI Bank 13031	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	2010	Nine Mile Point 2
Question Cognitive Level:	Memory or Fundamental Knowledge		X
	Comprehension or Analysis		
10 CFR Part 55 Content:	55.41 55.43		

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
	Topic and K/A #	G2	2.2.6
	Importance Rating		3.6

Equipment Control: Knowledge of the process for making changes to procedures.

Proposed Question: SRO Question # 95

Given the following:

- A surveillance is in progress on the Core Spray "A" system.
- The operator reports that the procedure cannot be completed as written because a procedure step mistakenly directs that the operator start Core Spray Pump "B".
- The surveillance will "drop dead" at the end of the shift.
- The Operations Department Manager is listed as the procedure owner.

IAW PNPS NOP98A1, PROCEDURE PROCESS and regarding the required procedure change:

The Emergent Work process (EWN) ______(1)_____ (can /cannot) be used.
The ______(2)_____ is the final authority in determining whether this process can be used

A. (1) Can
(2) The Operations Department Manager

B. (1) Cannot
(2) The Operations Department Manager

C. (1) Can
(2) The Shift Manager

D (1) Cannot
(2) The Shift Manager

Proposed Answer: C

A. Incorrect: Per page 16, the on-duty SM is responsible for concurring with the determination that emergent work is, in fact, emergent

B. Incorrect: The EWN process can be used if the change is a non-intent change and is associated with an emergent activity. Additionally, per page 16, the on-duty SM is responsible for concurring with the determination that emergent work is, in fact, emergent.

C. Correct: Per NOP98A1, the procedure change is a non-intent change. Non-intent changes include changing equipment identifiers (for example: changing B pump to A pump when the Procedure section obviously is written for the A pump). In order for the change to be processed using the EWN process, the change must be a non-intent change. In order to use the EWN process, the change must also be associated with an emergent activity.

Per page 11 of the NOP, deficiencies or conditions that mandate entry into a Technical Specifications Action statement having a 14-day or less shutdown requirement meets the criteria of emergent work. Per page 16, the on-duty SM is responsible for concurring with the determination that emergent work is, in fact, emergent

D. Incorrect: The EWN process can be used as the change is a non-intent change and is associated with an emergent activity as defined by the NOP

Technical Reference(s)	NOP98A1, Definitions [5] and [12] Page 16, and Attachment 6	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-06-06-01, EO-2b	(As available)

Question Source:	Bank	13720	
	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	
	55.43	3

Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		3
	Topic and K/A #	G3	2.3.11
	Importance Rating		4.3

Radiation Control: Ability to control radiation releases.

Proposed Question: SRO Question # 96

A large break LOCA occurred 24 hours ago. Drywell Hydrogen Concentration has been rising over the past 5 hours.

IAW EOP-03, which one of the following is correct regarding containment venting?

Venting is NOT permitted UNTIL H2 concentration is determined to be ...

A. ≥1%. It is not secured until H2 concentration is below 1% regardless of the offsite release.

B. ≥1%. It must be secured if the radioactivity release rate reaches the limits of the ODCM.

C. ≥4%. It is not secured until H2 concentration is below 4% regardless of the offsite release.

D ≥4%. It must be secured if the radioactivity release rate reaches the General Emergency Action levels.

Proposed Answer: B

A. Incorrect: Venting is only authorized if the release rate is expected to remain the limits of the ODCM as indicated by the Main Stack Lo Rad monitors reaching their Hi-Hi setpoints.

B. Correct: The H2 control leg of EOP-03 specifies that venting be commenced when H2 cannot be determined to be below 1%. The venting is contingent however on the release being below the limits of the ODCM. It goes on to direct that the vent is to be secured if the release reaches the ODCM limits.

C. Incorrect: Venting is commenced when H2 cannot be determined to be below 1%. Plausible in that 4% is also used in this leg of the EOP but is the level at which SAGs are entered. Additionally, venting is to be secured if the release reaches the ODCM limits.

D. Incorrect: Venting is commenced when H2 cannot be determined to be below 1%. Plausible in that 4% is also used in this leg of the EOP but is the level at which SAGs are entered. Additionally, venting is to be secured if the release reaches the ODCM limits. The release rates associated with a GE are significantly higher than the ODCM limits.

Technical Reference(s)	EOP-03 steps G-3 and G-4	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-02, EO-3i	(As available)
Question Source:	Bank	
	Modified Bank	(Note changes or attach parent)
	New X	
Question History:	Last NRC Exam: N/A	
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 #		4
	Topic and K/A #	G4	2.4.6
	Importance Rating		4.7

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies.

Proposed Question: SRO Question # 97

Given the following:

- EOP-18, Steam Cooling, is being executed following a loss of all injection sources
- Actual RPV water level is -155 inches and slowly lowering
- RPV Pressure is stable and being controlled in a band of 700 to 800 psig

Then,

- CRD is restored and injection commences
- Actual RPV water level is now -157 inches and still lowering

Which one of the following describes the required action and the reason for that action?

A. Exit EOP-18 and enter EOP-17, Emergency RPV Depressurization. The assumptions used while steam cooling without injection are no longer valid and adequate core cooling cannot be assured

B. Exit EOP-18, re-enter EOP-01 Pressure Leg and commence a controlled RPV depressurization until level can no longer be maintained above -160 inches. As long as level is above -160 inches, adequate core cooling is assured and lowering pressure will increase injection flow

C. Remain in EOP-18 but commence a controlled RPV depressurization until level can no longer be maintained above -160 inches. As long as level is above -160 inches, adequate core cooling is assured and lowering pressure will increase injection flow

D Remain in EOP-18 and continue efforts to align additional injection systems while maintaining pressure in band. Emergency Depressurizing now will result in an additional loss of inventory, shortening the time till adequate core cooling is lost.

Proposed Answer: A

A. Correct: The calculation for the MZIRWL (-160) assumes that there is no subcooling at the core inlet. Since any injection would invalidate this assumption, adequate core cooling cannot be assured if RPV water level is below the MSCRWL (-150) and water is being injected into the RPV. The 4th override in EOP-18, step S-1 directs that EOP-18 and EOP-17 be entered in this condition

B. Incorrect: An Emergency Depressurization is required. Plausible in that this would be the required action if level was rising

C. Incorrect: The assumptions for steam cooling without injection are no longer met and adequate core cooling may be lost. Emergency depressurization is required

D. Incorrect: Plausible in that this would be the action if CRD injection had not been restored

Technical Reference(s)	EOP-18 STEAM COOLING Lesson Plan O-RO-03-04-01, EOP-18, page 10	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-03-04-01, EO-3b	(As available)

Question Source:	Bank	PNPS Bank	
	Modified Bank		(Note changes or attach parent)
	New		
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 #		3
	Group #		3
	Topic and K/A #	G3	2.3.13
	Importance Rating		3.8

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: SRO Question # 98

Concerning the use of the Locked High Rad Area key controlled by the Shift Manager, which one of the following is correct regarding:

(1) The conditions that allow the use of this key AND

(2) If this key is used, requirements associated with entry into the Locked High Rad Area?

Assume the individual making the entry is a fully qualified non-licensed operator

A. (1) Any condition requiring entry into the Emergency Operating Procedures
(2) All normal entry and exit requirements may be bypassed.

B. (1) Any condition that requires entry into the Emergency Plan
(2) All normal entry and exit requirements may be bypassed.

C. (1) Any condition requiring entry into the Emergency Operating Procedures
(2) The shift manager is required to complete an RWP to the extent possible using the information available. All other normal entry and exit requirements may be bypassed.

D (1) Any condition that requires entry into the Emergency Plan
(2) The shift manager is required to complete an RWP to the extent possible using the information available. All other normal entry and exit requirements may be bypassed.

Proposed Answer: B

A. Incorrect: The key is to be used during conditions requiring entry into the Emergency Plan. Plausible in that most, but not all, EOP entry conditions require entry into the E-Plan (ex: low RPV level, high RPV pressure, area temp above MNOV)

B. Correct: The key is meant to be used during emergency conditions following entry into the E-Plan.

Plant personnel may bypass the normal RCA entry / exit process during emergencies. In such cases, all exposures should be recorded on an appropriate RWP as soon as possible after the event and a condition report generated to document the event.

C. Incorrect: The key is to be used during conditions requiring entry into the Emergency Plan. Additionally, Plant personnel may bypass the normal RCA entry / exit process during emergencies. In such cases, all exposures should be recorded on an appropriate RWP as soon as possible after the event and a condition report generated to document the event.

D. Incorrect: Plant personnel may bypass the normal RCA entry / exit process during emergencies. In such cases, all exposures should be recorded on an appropriate RWP as soon as possible after the event and a condition report generated to document the event.

Technical Reference(s)	EN-RP-101, section 5.1, items [11] and [12], section 5.11, item [12] and the associated note.	(Attach if not previously provided)

Proposed Reference to be provided to applicants during examination: None

Learning Objective:	O-RO-06-06-01, EO-95		(As available)
Question Source:	Bank		
	Modified Bank		(Note changes or attach parent)
	New	X	
Question History:	Last NRC Exam:	N/A	

Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
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 #		2
	Topic and K/A #	G2	2.2.5
	Importance Rating		3.2

Equipment Control: Knowledge of the process for making design or operating changes to the facility.

Proposed Question: SRO Question # 99

Given the following:

- The plant is operating at power.
- A transient has resulted in the performance of PNPS 2.2.42, Loss of RBCCW.
- The crew has just been directed by this procedure to crosstie the RBCCW Loops in accordance with Attachment 5 of this procedure.
- The SCRE states that this action meets the definition of a Temporary Modification.

Which ONE of the following is correct?

A. The SCRE is not correct and temporary modification controls of EN-DC-136, Temporary Modifications, are not needed.

B. The SCRE is correct however this action is specifically exempted from the Temporary Modification process.

C. The SCRE is not correct but this action must be documented via an eSOMS clearance after the plant is stable.

D The SCRE is correct, and the action must be documented IAW EN-DC-136, Temporary Modifications, after the plant is stable.

Proposed Answer: B

A. Incorrect: This is incorrect because the action meets the criteria for a TM. It is plausible because the operator may not know these criteria.

B. Correct: According to 3.0[25] of EN-DC-136 (p7), a Temporary Modification is a short-term alteration made to systems, structures, or components that do not conform to approved design configuration, unless excluded by this procedure. Crosstie of the RBCCW subsystems does not conform to the approved design configuration. Additionally, according to Attachment 9.2 of EN-DC-136 (p38-42; Rev9), there are 29 distinct exclusions from the Temporary Modification process, none of which are the action of cross-tying the RBCCW subsystems. Consequently, the SCRE is correct, in that the action does meet the criteria of a TM. However, according to a Note just prior to the actions of Attachment 5 of PNPS 2.4.42, due to the nature PNPS 2.4.42 and the required response time, the action of cross-tying the RBCCW subsystems is exempt from EN-DC-136, "Temporary Modifications".

C. Incorrect: This action does meet the definition of a Temp Mod.

D. Incorrect: This is incorrect because a Note in Attachment 5 of PNPS 2.4.42 specifically exempts the action from the TM process/procedure. Plausible since this does meet the definition of a Temp Mod.

Technical Reference(s)	EN-DC-136 (p7, 38-42; Rev 9) PNPS 2.4.42 Attachment 5	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-06-06-01 Objective 34a, 34b, and 34c	(As available)
Question Source:	Bank Modified Bank New	WTSI Bank (Note changes or attach parent)
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	 3

Facility licensee procedures required to obtain authority for design and operating changes in the facility.

Comments:

Examination Outline Cross-reference:		RO	SRO
	Level	RO	SRO
	Tier #		3
	Group #		4
	Topic and K/A #	G4	2.4.22
	Importance Rating		4.4

Emergency Procedures / Plan: Knowledge of the bases for prioritizing safety functions during abnormal/emergency operations.

Proposed Question: SRO Question # 100

The plant was at rated conditions when a fire in the main control room has necessitated evacuating the main control room.

IAW PNPS 2.4.143, Shutdown from Outside the Control Room, which one of the following correctly describes the initial priorities you would need to implement within the first 30 minutes?

A. Within 10 minutes transfer control of all four ADS valves to their Alternate Shutdown Panels to mitigate the consequences of spurious SRV operation. Within 24 minutes align an Injection source and commence injecting to mitigate the consequences of vessel boil-off.

B. Within 10 minutes align an Injection source and commence injecting to mitigate the consequences of vessel boil-off. Within 24 minutes transfer control of all four ADS valves to their Alternate Shutdown Panels to mitigate the consequences of spurious SRV operation.

C. Within 10 minutes transfer control of all four ADS valves to their Alternate Shutdown Panels to mitigate the consequences of spurious SRV operation. Within 24 minutes align RHR for torus cooling in order to maintain torus temperature below the Heat Capacity Temperature Limit over the duration of the event.

D Within 10 minutes align an Injection source and commence injecting to mitigate the consequences of vessel boil-off. Within 24 minutes align RHR for torus cooling in order to maintain torus temperature below the Heat Capacity Temperature Limit over the duration of the event.

Proposed Answer: A

A. Correct: Per PNPS 2.4.143 discussion section, item [6], SRV ASPs need to be transferred within 10 minutes and an injection system lined up within 24 minutes.

B. Incorrect: Per PNPS 2.4.143 discussion section, item [6], SRV ASPs need to be transferred within 10 minutes and an injection system lined up within 24 minutes.

C. Incorrect: Engineering analysis indicates that torus cooling does not need to be aligned until the 2 hour point.

D. Incorrect: Engineering analysis indicates that torus cooling does not need to be aligned until the 2 hour point.

Technical Reference(s)	PNPS 2.4.143 discussion section, item [6]	(Attach if not previously provided)
Proposed Reference to be provided to applicants during examination:		None
Learning Objective:	O-RO-04, Task 200-05-01-014	(As available)

Question Source:	Bank	PNPS Bank	
	Modified Bank		(Note changes or attach parent)
	New		
Question History:	Last NRC Exam:	Not used	

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: