

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000	K1.02
	Importance Rating	2.7	

Knowledge of the connections and / or cause effect relationships between INSTRUMENT AIR SYSTEM and the following: Service air

Proposed Question: RO Question # 1

Given the following:

- Low pressure service air is being supplied via the HI/LOW Service Air cross connect
- An air leak occurs at some unknown location
- The K-117 Air Compressor starts and stabilizes air system pressure at 78 psi.

Assuming that NO operator action has been taken:

(1) What is the position of AO-4350, Serv Air Hdr Block Valve

AND

(2) AO-4353, High Press/Low Press Service Air Inlet Crosstie Valve?

	<u>AO-4350, Serv Air Hdr Block Valve</u>	<u>AO-4353, High Press/Low Press Service Air Inlet Crosstie Valve</u>
A.	Open	Open
B.	Open	Closed
C.	Closed	Open
D.	Closed	Closed

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: AO-4353 closes at 85 psig. Plausible in that AO-4365, Non-Ess Instr Air Hdr Blk Vlv, closes at 80 psig.
- B. Incorrect: AO-4350 closes at 85 psig. Plausible in that AO-4365, Non-Ess Instr Air Hdr Blk Vlv, closes at 80 psig.

- C. Incorrect: Both valves close at 85 psig. Plausible in that AO-4365, Non-Ess Instr Air Hdr Blk Vlv, closes at 80 psig.
- D. Correct: Per PNPS 5.3.8, at 85 psig AO-4350, Serv Air Hdr Block Vlv, closes to isolate the service air header and AO-4353, High Press/Low Press Service Air Inlet Crosstie Valve, closes to isolate instrument air from low pressure service air header.

Technical Reference(s): PNPS 5.3.8, pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-02-04, EO-9 (As available)

Question Source: Bank # WTSI Bank 12738  
 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	
	K/A #	262002	K1.19
	Importance Rating	2.9	

Knowledge of the physical connections and/or cause- effect relationships between UNINTERRUPTABLE POWER SUPPLY (A.C./D.C.) and the following: Power range neutron monitoring system: Plant-Specific  
Proposed Question: RO Question # 2

A loss of 120 VAC Vital Bus Y-2 has occurred. Which one of the following is correct regarding the ability to monitor reactor power on Panel C905?

- A. APRM and IRM recorder indications are lost  
APRM, IRM and SRM status lights are lost
- B. APRM and IRM recorder indications remain functional  
APRM, IRM and SRM status lights remain functional
- C. APRM and IRM recorder indications are lost  
APRM, IRM and SRM status lights remain functional
- D. APRM and IRM recorder indications remain functional  
APRM, IRM and SRM status lights are lost

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: APRM, IRM and SRM status lights are not lost. Plausible in that these are lost if a loss of 120 VAC Bus Y-1 has occurred.
- B. Incorrect: APRM and IRM recorder indications are lost. Plausible in that they will remain functioning if a loss of Instrument bus Y-1 occurs.
- C. Correct: APRM/IRM recorder indication is lost.

Also IAW section 5.0 PNPS 5.3.6, Discussion Section item 1, the indications below are still available on C905 panel:

- All SRM/IRM/APRM/RBM status lights on the C905 horizontal section

D. Incorrect: APRM and IRM recorder indications are lost. Plausible in that they will remain functioning if a loss of Instrument bus Y-1 occurs.

Technical Reference(s): PNPS 5.3.7, Discussion Section 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-01-07, EO 5b (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 X  
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000	K2.02
	Importance Rating	2.5	

Knowledge of electrical power supplies to the following: Valves (RHR/LPCI: Injection Mode)

Proposed Question: RO Question # 3

The 480 VAC distribution is aligned for normal full power operation when 480 VAC Load Center B-1 trips.

The status of the B-1 to B-6 supply breakers is as follows:

- Breaker 52-102 on B-1 is closed (B-6 Supply Breaker)
- Breaker 52-601 on B-6 is open (B-6 Supply Breaker)

What is the status of power availability to the following valves?

- LPCI Injection Valves #2, MO-1001-29A and B
- LPCI Injection Valves #1, MO-1001-28A and B
- Recirc Pump Discharge Valves, MO-202-5A and B

- A. All valves have power
- B. NO valves have power
- C. MO-1001-29B has power  
MO-1001-28B has power  
MO-202-5B has power  
All other valves do NOT have power
- D. MO-1001-29A and B have power  
MO-1001-28A and B have power  
MO-202-5B has power  
MO-202-5A does NOT have power

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Bus B-6 is de-energized. Swing bus B-6 will not transfer to B-2 unless both

breakers from B-1 are open. Bus B-6 supplies all valves that must re-position on a LPCI loop selection via B-20. All of the noted valves may be required to reposition and are therefore are without power. Plausible if the candidate thinks that with the one supply breaker open, B-6 will transfer.

- B. Correct: 480 V LC B-6 can be energized from B-1 (normal) or B-2 (alternate). Normally if bus B-1 suffers loss of voltage for > one second, 52-102 and 52-601 will trip open and breakers 52-202 and 52-602 automatically close. However if the B-1 supply, 52-102 fails to open the corresponding supply from B-2, 52-202 will be interlocked open. Therefore B-6 will not auto transfer. Bus B-6 supplies all valves that must re-position on a LPCI loop selection via B-20. All of the noted valves may be required to reposition and are therefore without power.
- C. Incorrect: All valves are without power. Plausible if the candidate thinks the "B" side valves are powered from the "B" side of the electric plant and not the swing bus.
- D. Incorrect: All valves are without power. Plausible if the candidate does not realize that the Recirc valves that must reposition on a LPCI loop select are also powered from swing bus B-6 and concludes that the "B" Recirc discharge valve is still has power available.

Technical Reference(s): 480 VAC Reference Text, pages (Attach if not previously provided) 16, 39, 40, 41 and Figure 2

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-01, EO 15a (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	
	K/A #	215003	K2.01
	Importance Rating	2.5	

Knowledge of electrical power supplies to the following: IRM channels/detectors

Proposed Question: RO Question # 4

IRM Channel "C" has just gone INOP.

Which one of the following electrical losses could have resulted in this occurrence?

- A. 24 VDC D-25 trip
- B. 24 VDC D-26 trip
- C. 120 VAC Instrument Bus Y-1 trip
- D. 120 VAC Safeguard Power Supply Y-3 trip

Proposed Answer: A

Explanation (Optional):

- A. Correct: D-25 supplies IRM Channels A, C, E, and G. An INOP trip is generated on loss of high voltage power supply.
- B. Incorrect: D-26 supplies IRM Channels B, D, F, and H.
- C. Incorrect: Y-1 supplies the detector drive.
- D. Incorrect: Y-3 does not supply any component within the IRM system.

Technical Reference(s): 24 VDC Distribution Reference (Attach if not previously provided)  
Text, page 7  
IRM Reference Text, page 18  
5.3.7, Discussion, [5]

Proposed References to be provided to applicants during examination: None



Learning Objective:

(As available)

Question Source: Bank #

Modified Bank # WTSI 1274

Modified for Pilgrim and changed stem to say INOP trip vice just loss of power.

New

Question History:

Last NRC Exam: 2007 Fermi

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	211000	K3.01
	Importance Rating	4.3	

Knowledge of the effect that a loss or malfunction of the STANDBY LIQUID CONTROL SYSTEM will have on following: Ability to shutdown the reactor in certain conditions.

Proposed Question: RO Question # 5

During an ATWS, Standby Liquid Control (SLC) injection is required. The following sequence occurs:

- The operator places SLC ACTUATE switch to "SYS A"
- SLC Pump "A" starts
- SLC System "A" squib valve fails to fire
- The operator places SLC ACTUATE switch to "SYS B"
- SLC Pump "B" starts and injects
- The breaker for SLC Pump "B" trips 15 seconds later

Which one of the following is correct regarding any required actions to shutdown the reactor using boron injection?

- A. Boron can only be injected using RWCU.
- B. The SLC ACTUATE switch should be switched to "SYS A".
- C. The SLS System "A" local pushbutton must be depressed.
- D. Attempt one reset of the SLC Pump "B" breaker. If the breaker trips again, inject with RWCU.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Boron will inject if the SLC ACTUATE switch is placed to "A".
- B. Correct: The discharge of the two pumps are cross-tied. Either pump will inject thru either squib valve. Although the "A" squib failed to fire, the "A" pump started. When SYS "B" was started the pump was injecting thru the "B" squib valve. After the "B" pump tripped, if the switch is placed back in SYS "A", the "A" pump will restart and inject thru the "B" squib.

- C. Incorrect: The local pushbutton will only start the pump; it will not fire the squib valve.
- D. Incorrect: Injection capability is available from the control room.

Technical Reference(s): 2.2.24, Sect. 7.2 (Attach if not previously provided)  
 SLC Reference Text, Figure 1  
 SLC Reference Text, local controls

Proposed References to be provided to applicants during examination:

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

Question Source: Bank # PNPS 15625  
 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	
	K/A #	215005	K3.07
	Importance Rating	3.2	

Knowledge of the effect that a loss or malfunction of the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM will have on following: Rod Block Monitor

Proposed Question: RO Question # 6

A power ascension with control rods is in progress. Current reactor power is 52% and a center control rod is being withdrawn. During the rod withdraw, APRM channel "B" fails downscale.

Assuming no operator actions, which one of the following describes the response of the Rod Block Monitoring System?

- A. A RBM INOP trip is generated
- B. A RBM Downscale trip is generated
- C. An automatic bypass of RBM channel "B" occurs
- D. The reference APRM signal to RBM "B" automatically shifts to APRM "D"

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The RBM automatically bypasses. Plausible if the candidate knows that the RBM uses the APRMs and concludes that the reference APRM failure will cause a RBM Inop trip.
- B. Incorrect: The RBM automatically bypasses. Plausible if the candidate recalls that the APRM is the reference APRM for RBM "A" and concludes that the reference APRM is used for determining that the RBM is downscale vice no longer required.
- C. Correct: APRM "B" is the "Normal" reference to RBM "A". When the APRM fails downscale, the RBM responds as if power is < 26%. The RBM is not required below 26% and an automatic RBM bypass occurs.

D. Incorrect: The RBM automatically bypasses. Plausible if the candidate recalls that there is a backup reference APRM for each RBM channel. However the RBM will not shift to its backup reference until the "Normal" APRM is bypassed. APRM "D" is the backup for RBM "B".

Technical Reference(s): RBM Reference Text, pages 11, (Attach if not previously provided)  
12

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-05, EO 13b (As available)

Question Source: Bank # 227  
Modified Bank # (Note changes or attach parent)  
New

Question History: Last NRC Exam: 2009 Pilgrim, Question 6

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 #	1	
	K/A #	206000	K4.14
	Importance Rating	3.4	

Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following: Control oil to turbine speed controls:

Proposed Question: RO Question # 7

HPCI operation is required to maintain adequate core cooling. However rated injection flow is not required.

Which one of the following is a system operating limit that ensures that there is adequate control oil pressure to the turbine speed controls?

Maintaining HPCI ....

- A. Injection  $\geq$  1000 gpm
- B. Injection  $\geq$  3000 gpm
- C. Turbine speed  $\geq$  1000 rpm
- D. Turbine speed  $\geq$  3000 rpm

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The limit is 1000 rpm on the turbine. Plausible if the candidate confuses the RPM limit as an injection limit as there is a limit on gpm injection (see below.) However this limit applies to automatic flow control.
- B. Incorrect: The limit is 1000 rpm on the turbine. Plausible in that this is the limit for operating the HPCI turbine in automatic flow control.
- C. Correct: IAW PNPS 2.2.21.5, Attachment 1, Section 4.0 Caution, during normal operation, the HPCI turbine should not be run below 2000 RPM. Below 2000 RPM, intermittent exhaust flow will cause water hammer in the exhaust line. If HPCI turbine operation below 2000 RPM is required to achieve and/or maintain adequate core cooling, then the HPCI turbine should not be run below 1000 RPM. This will ensure adequate oil pressure to the control oil system and bearing lubrication.

D. Incorrect: The limit is 1000 RPM. Plausible in that there is a limit associated with the number 3000 but it is associated with injection flow and has to do with controller operations.

Technical Reference(s): 2.2.21.5, Attachment 1, Section 4.0 Caution (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-03, EO # 11 (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 8  
55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001	K4.02
	Importance Rating	3.0	

Knowledge of LOW PRESSURE CORE SPRAY SYSTEM design feature(s) and/or interlocks which provide for the following: Prevents water hammer

Proposed Question: RO Question # 8

Core Spray is aligned for normal full power operation.

Which one of the following is correct regarding the prevention of water hammer in the core spray system?

- A. Condensate Transfer system pumps maintain both the Core Spray pump suction and discharge piping full of water.
- B. Condensate Transfer system pumps maintain the Core Spray pump discharge piping full of water. Torus water maintains the Core Spray pump full.
- C. Demin Water Transfer system pumps maintain both the Core Spray pump suction and discharge piping full of water.
- D. Demin Water Transfer system pumps maintain the Core Spray pump discharge to keep the discharge piping full of water. Torus water maintains the Core Spray pump full.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Torus water maintains the pump suction and pump piping full of water. Plausible if the candidate believes that Condensate Transfer is supplied to the Core Spray pump suction. This is plausible if the candidate recalls that Condensate transfer water is supplied to the RHR pump suction line (for flushing purposes).
- B. Correct: Water from the Condensate Transfer System pumps (Transfer or Jockey pumps) is supplied to the Core Spray discharge /injection line downstream of the pump discharge check valve. This "Keep-fill" maintains the piping pressure at 110 psig to prevent water hammer and piping damage when the CS pumps start.

Torus water maintains the pump suction and pump full of water up to the pump discharge valve. Condensate Transfer pump discharge pressure is much higher than



the static pressure of the torus and the Core Spray pump discharge check valve is kept closed.

- C. Incorrect: Condensate transfer supplies the water. Plausible in that Demin water provides makeup to various systems.
- D. Incorrect: Condensate transfer supplies the water. Plausible in that Demin water provides makeup to various systems.

Technical Reference(s): Core Spray reference text, page 14 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

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

Question Source: Bank #  
Modified Bank # PNPS Bank # 2875 Modified stem to address core spray vice RHR. Also modified to require candidate to how each section of piping is maintained full of water.  
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 8  
55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002	K5.01
	Importance Rating	3.1	

Knowledge of the operational implications of the following concepts as they apply to REACTOR WATER LEVEL CONTROL SYSTEM : GEMAC/Foxboro/Bailey controller operation: Plant-Specific

Proposed Question: RO Question # 9

A reactor plant shutdown is in progress. The "B" Feed Reg Valve was placed in manual as power was lowered to steady out observed water level control oscillations.

Shortly after commencing the down power...

- A Recirc pump trips.
- Reactor power stabilizes at 65%.
- RPV water level stabilizes at 30 inches
- FWLC individual M/A station's are currently indicating as shown in the picture to the right



Given these indications and assuming the power reduction continues, which one of the following is correct regarding:

- (1) The response of the controller to the Recirc pump trip AND
  - (2) Any require action?
- A. (1) The "A" M/A station is NOT responding properly. If not corrected a low vessel level scram will occur.  
(2) Place "A" FRV in manual, increase the output signal to balance flows.
  - B. (1) The "A" M/A station IS responding properly. If not corrected a turbine trip on high level will occur.  
(2) Lower the output signal of the "B" FRV at the "B" M/A station.
  - C. (1) The "B" FRV is NOT responding properly. If not corrected a turbine trip on high

- level will occur.
- (2) Lock the FRV in the condenser compartment and attempt to throttle the valve closed to balance flows
- D.
- (1) The "B" M/A station IS responding properly. If not corrected a low vessel level scram will occur.
  - (2) Raise the Setpoint on the Master Controller to open the "A" FRV further.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The "A" controller is responding correctly in Auto. The "B" controller being in manual, maintained the "B" FRV at the position required for high power operation. After the Recirc pump tripped, level rose due to the amount of feed going thru the "B" Recirc pump. The "A" controller cut back the "A" FRV in response. Going to manual and raising the feed thru "A" FRV will only cause level to rise.
- B. Correct: The "A" controller is responding correctly in Auto. The "B" controller being in manual, maintained the "B" FRV at the position required for high power operation. After the Recirc pump tripped, level rose due to the amount of feed going thru the "B" Recirc pump. The "A" controller cut back the "A" FRV in response. If the down power continues the "A" FRV will go full close. Any additional reduction will result in a rising level and turbine trip. The required action is to lower the flow thru the "B" FRV which will cause the "A" FRV to open further to maintain level providing more margin to the valve going close and will balance feed flows.
- C. Incorrect: The "B" FRV is responding properly. Its output is higher than "A"'s only because it was in manual and set to maintain level at the initial high power condition. Plausible if the candidate focuses on the output signal and concludes that the controller has failed. In that case the action described would be correct.
- D. Incorrect: If the power reduction is continued, the "A" FRV will go close, level will begin to rise and eventually reach the high level turbine trip. Raising the Auto setpoint will only cause a momentary opening of the "A" FRV which will eventually return to its original position.

Technical Reference(s): FWLC Reference Text, page 18 & (Attach if not previously provided) 19 and figure 1a and 9

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-04-10, EO 13 (As available)

Question Source: Bank # LOR Bank # 43  
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	
	K/A #	215004	K5.03
	Importance Rating	2.8	

Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM : Changing detector position

Proposed Question: RO Question # 10

A reactor startup is in progress. The Reactor Mode Switch is in STARTUP and all IRMs are on range 2

SRM Channel "A" is selected for withdraw and the Withdraw pushbutton has just been depressed.

Which one of the following will be the FIRST to occur as the detector is withdrawn?

- A. SRM Downscale rod block
- B. SRM Inoperable rod block
- C. Detector retract not permitted rod block
- D. Automatic termination of the detector withdraw

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The SRM downscale will not occur until count rate drops to 3 CPS.
- B. Incorrect: An SRM inoperable trip is caused by:
  - Low detector voltage
  - SRM module unplugged
  - The SRM function switch is not in OPERATE
- C. Correct: If any SRM channel detects < 100 cps and its detector is not fully inserted in the core, a rod block is activated. This rod block is bypassed if:
  - The mode switch is in RUN
  - The associated IRM range switches are at range three or above (IRM channels A, C, E, and G range switches bypass SRM A and C, IRM channels B, D, F, and

H bypasses SRM B and D).  
Since the IRMs are on range 2, this rod block will occur.

- D. Incorrect: Detector will not automatically stop. Plausible if the candidate concludes that this action will occur to prevent the downscale rod block at 3 CPS.

Technical Reference(s): SRM Reference Text, pages 17 (Attach if not previously provided)  
and 18

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-01, EO 6 (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	
	K/A #	264000	K6.08
	Importance Rating	3.6	

Knowledge of the effect that a loss or malfunction of the following will have on the EMERGENCY GENERATORS (DIESEL/JET) : A.C. power

Proposed Question: RO Question # 11

The startup transformer is supplying all 4160 VAC buses.

Then the following occurs:

- Due to problems on the 345 KV electrical system, voltage on Pilgrim's 4160 VAC distribution begins to lower.
- Bus undervoltage alarms for both A5 and A6 buses annunciate.
- Bus voltage drops to 3810 volts and stabilizes at this value.

What automatic actions should take place for this occurrence?

Startup transformer breakers for A5 and A6 will ....

- trip after approximately 1.2 seconds. Diesel generators will start and power A5 and A6.
- trip after approximately 10 seconds. Diesel generators will start and power A5 and A6.
- remain closed but load shed will initiate after approximately 10 seconds.
- trip after approximately 10 seconds. Diesel generators will not start but A5 and A6 will be powered by the shutdown transformer.

Proposed Answer: B

Explanation (Optional):

- Incorrect: Breakers will not trip for 10 seconds. Plausible in that there is a 1.2 breaker trip but that will not occur unless voltage drops to < 3780 volts.
- Correct. If the Startup Transformer feeder breaker to A5 or A6 is CLOSED, then the breaker will trip when Startup Transformer secondary voltage is less than or equal to 3858.92 to 3898.48V after a 9.88 to 10.60-second time delay. The diesels will start if:

- The Aux Trans breaker is open AND
  - The Startup transformer (S/U Xfmr) breaker is open AND
  - Degraded S/U Xfmr. Secondary Volts, 3880 volts for  $\geq 10$  seconds
- So when the S/U transformer breaker was tripped open by the degraded voltage condition, the diesels started and re-energized their buses.

- C. Incorrect: Load shed will not initiate because an accident signal is not present. Plausible in that the undervoltage alarms input into the load shed circuit and if the candidate believes that load shed is initiated in an attempt to recover voltage. However this will only occur if an ECCS start signal was also present.
- D. Incorrect: The diesels will re-power the buses. Plausible if the candidate does not think the EDGs will start at this voltage level. If so, then the shutdown transformers would energize the buses.

Technical Reference(s): Emergency AC Reference test, (Attach if not previously provided)  
page 18

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-08 EO-4 (As available)

Question Source: Bank # 14153 Made editorial changes to stem and replaced distractor "A"  
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 8  
55.43

Components, capacity, and functions of emergency systems.

Comments:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	217000	K6.01
	Importance Rating	3.4	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC): Electrical power

Proposed Question: RO Question # 12

The plant is at rated conditions with RCIC in its normal standby lineup.

Which of the following power losses will prevent RCIC from auto starting and injecting at rated flow if the initiation setpoint was reached?

- Loss of 480 VAC MCC B-18
  - Loss of 125 VDC Bus D-6
  - Loss of 125 VDC Bus D-16
  - Loss of 125 VDC Bus D-17
- A. loss of 125 VDC Bus D-16 only
- B. loss of 125 VDC Bus D-17 only
- C. loss of 125 VDC Bus D-6 or the loss of 480VAC MCC B-18
- D. loss of 125 VDC Bus D-16 or the loss of 480VAC MCC B-18

Proposed Answer: A

Explanation (Optional):

- A. Correct: A loss of D-16 will prevent RCIC from starting as all motor operated valves and control circuitry except for MO-1301-16 from power center D-16. D-16 supplies D4 and D7. D4 supplies the RCIC inverter, and D-7 supplies the DC MOVs.
- B. Incorrect: A loss of D-17 will not prevent RCIC from starting. Plausible in that it will prevent HPCI from starting.
- C. Incorrect: Neither a loss of D-6 or B-18 will prevent RCIC from starting. Loss of B-18 will not prevent RCIC from auto starting. Plausible in that B-18 powers Steam Supply Valve MO-1301-16. However this valve is normally open when RCIC is in its standby lineup.

D. Incorrect: Loss of B-18 will not prevent RCIC from auto starting. Plausible in that B-18 powers Steam Supply Valve MO-1301-16. However this valve is normally open when RCIC is in its standby lineup.

Technical Reference(s): PNPS 5.3.11 (Attach if not previously provided)

PNPS 5.3.12

125 VDC Reference Text, page 30

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-04, EO 13 (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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002	A1.08
	Importance Rating	3.8	

Ability to predict and/or monitor changes in parameters associated with operating the RELIEF/SAFETY VALVES controls including: Suppression pool water temperature  
Proposed Question: RO Question # 13

The plant is starting up from an outage that included replacing the SRVs.

Testing of the new SRV is in progress.

IAW Tech Specs which one of the following is the LOWEST Torus water temperature which requires that the SRV testing be secured?

- A. 80°F
- B. 90°F
- C. 110°F
- D. 120°F

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Testing can continue up to 90°F. Plausible in that 80°F is the maximum temperature during normal continuous operation IAW TS 3.7.A.1.c. However during testing the limit is raised to 90 degrees.
- B. Correct: IAW TS 3.7.A.1.d, the maximum suppression pool bulk temperature during RCIC, HPCI or ADS operation shall be ≤ 90°F. TS 3.7.A.1.f goes on to say that If the suppression pool bulk temperature exceeds the limits of Specification 3.7.A.1 .d, RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.
- C. Incorrect: Testing must be secured when torus temperature exceeds 90 degrees. Plausible in that this is the TS limit at which a scram is required as required by TS 3.7.A.1.g.

- D. Incorrect: Testing must be secured when torus temperature exceeds 90 degrees. Plausible in that this is the TS limit at which the vessel must be depressurized to  $\leq 200$  psig as required by TS 3.7.A.1.g.

Technical Reference(s): TS 3.7.A.1.c, d, f, g (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-03, EO 15g (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 X  
Comprehension or Analysis

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 #	2	
	Group #	1	
	K/A #	205000	A1.05
	Importance Rating	3.4	

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

Reactor water level

Proposed Question: RO Question # 14

Shutdown Cooling is being placed in service and the RHR pump is about to be started.

IAW PNPS 2.2.19.1, Shutdown Cooling Operations, which one of the following is correct regarding:

(1) the indicated RPV level response when the RHR pump is first started

AND

(2) the required action taken before starting the RHR pump in anticipation of that indicated level response?

- A. (1) Level will lower  
(2) Raise level to > +60 inches
- B. (1) Level will lower  
(2) Raise level to > +30 inches
- C. (1) Level will rise  
(2) Lower level to between +20 and +25 inches
- D. (1) Level will rise  
(2) Align RWCU for reject operation

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The required action is to raise level to > +30 inches. Plausible if the candidate confuses the water level requirement for a loss of SDC cooling (Raise RPV

level to > +60 inches).

- B. Correct: RPV level will lower when the pump is started. PNPS 2.2.19.1 directs that RPV level be raised to > +30 inches in order to prevent a low RPV level SDC isolation when the pump is started.
- C. Incorrect: RPV level will lower when the pump is started. Plausible if the candidate confuses the plant response with what occurs when securing the RHR pump. If so the candidate would apply the required action prior to securing the pump (lower level to between +20 and +25 inches).
- D. Incorrect: RPV level will lower when the pump is started. Plausible if the candidate confuses the plant response with what occurs when securing the RHR pump. Although plausible, the required action is not to verify or align RWCU for reject operation.

Technical Reference(s): PNPS 2.2.19.1, page 46 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-01, TO, 205-01-01-018 (As available)  
PLACE RHR IN SHUTDOWN  
COOLING.

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 #	2	
	Group #	1	
	K/A #	223002	A2.10
	Importance Rating	3.9	

Ability to (a) predict the impacts of the following on the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Loss of coolant accidents

Proposed Question: RO Question # 15

The plant was at rated conditions when a loss of coolant accident occurred. Current plant conditions are as follows:

- The Reactor Mode Switch is in Shutdown
- Drywell pressure is 12 psig and rising
- RPV level is + 58 inches and slowly rising due to swell
- RPV Pressure is 700 psig and slowly rising
- Main Steam Tunnel temperature is 160 °F and rising slowly due to a loss of ventilation
- HPCI injected but is now tripped on high level
- The MSIVs have automatically closed
- EOPs require that the MSIVs be reopened in order to perform an Emergency Depressurization using the bypass valves

Which one of the following is correct regarding:

- The Group 1 MSIV isolations signals that are currently present
- AND
- IAW PNPS 5.3.21, the actions directed to defeat the isolation signal(s) in order to reopen the MSIVs?

- A. (1) High RPV Level only  
(2) The PCIS RX WTR LVL Bypass switches on Panels C915 and C917 must be placed in BYPASS
- B. (1) High RPV Level only  
(2) Jumpers must be installed on Panels C915 and C917
- C. (1) High Main Steam Tunnel Temperature only  
(2) Jumpers must be installed on Panels C915 and C917

- D. (1) High RPV Level and high Main Steam Tunnel Temperature  
(2) The PCIS RX WTR LVL Bypass switches on Panels C915 and C917 must be placed in BYPASS AND Jumpers must be installed on Panels C915 and C917

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Jumpers must be installed to defeat this isolation. The PCIS Bypass Switches only defeat the Low RPV level isolation (-46 inches). Therefore jumpers must be installed IAW Attachment 2 of PNPS 5.3.21. Plausible due to the name of the switches and the candidate believes that both water level isolation signals are bypassed by these switches.
- B. Correct: An RPV High Water level isolation signal is present due to RPV level being > +55 inches, Mode Switch out of Run with a low pressure condition. IAW Attachment 2 of PNPS 5.3.21, jumpers must be installed to defeat this isolation.
- C. Incorrect: A Main Steam Tunnel Isolation does not exist and a high level isolation does exist. Plausible in that the steam tunnel temperature is above the isolation setpoint for the Turbine basement Exhaust (155 °F) but not the Steam Tunnel setpoint of 175 °F. Additionally, it also plausible that the candidate may not recognize the high level isolation as this isolation signal is automatically bypassed under specified conditions.
- D. Incorrect: A Main Steam Tunnel Isolation does not exist and a high level isolation does exist. Plausible in that the steam tunnel temperature is above the isolation setpoint for the Turbine basement Exhaust (155 °F) but not the Steam Tunnel setpoint of 175 °F. The required action is plausible if the candidate believes that a high temperature isolation exists and that the PCIS Water level bypass switches bypass the high level also.

Technical Reference(s): PNPS 2.2.125, Attachment 3 (Attach if not previously provided)  
PNPS 5.3.21, Attachments 1 and 2.

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-10, EO 3, and 5 (As available)

Question Source: Bank #  
Modified Bank #



New X

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 7  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001	A2.04
	Importance Rating	3.8	

Ability to (a) predict the impacts of the following on the A.C. ELECTRICAL DISTRIBUTION; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Types of loads that, if de-energized, would degrade or hinder plant operation.

Proposed Question: RO Question # 16

The plant is at rated conditions when the preferred Main Transformer Cooling power supply from B-4 is lost.

- (1) Which one of the following is correct regarding the Backup/Emergency power supply to the Main Transformer Coolers,

AND

- (2) What action would be required if this the Backup/Emergency power supply was unavailable?

- A. (1) The power supply to the coolers will automatically shift to the Backup/Emergency source from B-3.  
(2) Reactor power would have to be reduced below the Main Transformer's self-cooled rating.
- B. (1) The power supply to the coolers will automatically shift to the Backup/Emergency source from B-3.  
(2) The Main Transformer would have to be removed from service.
- C. (1) The power supply to the coolers must be manually shifted to the Backup/Emergency source from B-3.  
(2) Reactor power would have to be reduced below the Main Transformer's self-cooled rating.
- D. (1) The power supply to the coolers must be manually shifted to the Backup/Emergency source from B-3.  
(2) The Main Transformer would have to be removed from service.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The transformer would have to be removed from service as there is no self-cooled rating for the Main. Plausible in that both the Startup and Unit Aux transformers have self-cooled ratings.
- B. Correct: The transformer cooling fans and oil pumps are fed from either of two station 480 VAC load centers. The normal source is from bus B-4 and the alternate/emergency source is from bus B-3. An automatic transfer scheme is provided to re-energize the coolers from the alternate/emergency feeder should the normal source voltage fail. Since the transformer does not have a self-cooled rating, it must be removed from service.
- C. Incorrect: The power supply shifts automatically to the backup power supply. Additionally, the transformer must be removed from service.
- D. Incorrect: The power supply shifts automatically to the backup power supply.

Technical Reference(s): Main Generator reference text, (Attach if not previously provided)  
pages 18 and 19  
PNPS 2.2.2 page 11

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-01-07, EO-5 (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 4  
55.43

Secondary coolant and auxiliary systems that affect the facility.  
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000	A3.01
	Importance Rating	3.0	

Ability to monitor automatic operations of the CCWS including: Setpoints on instrument signal levels for normal operations, warnings, and trips that are applicable to the CCWS

Proposed Question: RO Question # 17

A small steam leak in the drywell results in drywell pressure rising to 3.0 psig.

The reactor scrams but the Startup Transformer locks out when house loads attempt to transfer. Current electric plant conditions are as follows:

- The "A" EDG failed to start
- 4160 VAC bus A5 energized by the Shutdown Transformer
- 4160 VAC bus A6 energized by the "B" EDG

Which one the following describes impact of the above on the TBCCW system?

- A. Neither TBCCW pump is available due to the load shed signals.
- B. The "A" TBCCW pump will auto start on low pressure 20 seconds after TBCCW pressure was lost.  
The "B" TBCCW pump is not available due to the load shed signal.
- C. The "B" TBCCW pump will be inhibited from starting until all ECCS pumps are running and then will restart on low system pressure.  
The "A" TBCCW pump is not available due to the load shed signal.
- D. The "A" TBCCW pump will be inhibited from starting until all ECCS pumps are running and then will restart on low system.  
The "B" TBCCW pump is available for auto start if the "A" TBCCW pump fails.

Proposed Answer: A

Explanation (Optional):

- A. Correct: ER02116887 modified the load shed logic for TBCCW Pumps A and B (P-110A and P-110B). This modification changed the load classification of TBCCW Pumps A and B (P-110A and P-110B) from a Group 3 to a Group 2 load shed signal. Following the loss of 345kV offsite AC power or a degraded voltage on the Startup Transformer, coincident with a LOCA initiation signal, P-110A and P-110B will not auto-start. Diesel

Load Shedding was initiated on A6 when the "B" EDG powered the bus in conjunction with the high drywell pressure condition. Diesel Load Shedding was initiated on A5 when due to the SUT supply breaker being open, the UAT supply open and a high drywell pressure signal.

- B. Incorrect: Neither pump is available. Plausible if the candidate does not realize that the SDT powering the bus would also result in an EDG load shed signal.
- C. Incorrect: Neither pump is available. Plausible if the candidate does thinks the TBCCW pumps are Group 3 loads (as they were in the past) and restart following a time delay to allow the LP ECCS pumps to start.
- D. Incorrect: Neither pump is available. Plausible if the candidate does not understand that a load shed condition is present on both buses.

Technical Reference(s): 2.4.41, Discussion item #5 (Attach if not previously provided)  
Figure 12 of Emergency AC reference Text.

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: 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	
	K/A #	261000	A3.03
	Importance Rating	3.0	

Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM

including: Valve operation

Proposed Question: RO Question # 18

With the "A" Standby Gas Treatment train in its normal standby lineup, an operator places the control switch for the "A" SBGT fan to the RUN position.

Which one of the following is correct?

The "A" SBGT fan will ...

- A. start and its outlet and inlet dampers will open.
- B. not start until the operator manually opens the outlet and inlet dampers.
- C. start, the outlet damper will open, and a low flow alarm will result due to the inlet damper not being opened by the operator.
- D. not start until the operator establishes a complete suction flow path to the fan from the contaminated exhaust duct or from the drywell or torus.

Proposed Answer: A

Explanation (Optional):

- A. Correct: The fan will start. After the fan starts, the inlet and outlet dampers will open.
- B. Incorrect: The fan will start. Plausible in that the "B" train is started by first opening the inlet damper which will in turn start the "B" fan. The "B" fan starting will then open the outlet damper.
- C. Incorrect: The inlet damper will open. Plausible if the candidate does not understand that the fan starting will open the inlet damper.
- D. Incorrect: The fan will start as described in "A" and both the inlet and outlet dampers will open. There is no interlock preventing system start if a complete suction path is not first started.

Technical Reference(s): SBT Reference Text ,page 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-03, EO 10 (As available)

Question Source: Bank # PNPS 3929  
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 7  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000	A4.05
	Importance Rating	4.3	

Ability to manually operate and/or monitor in the control room: Reactor power (RPS)

Proposed Question: RO Question # 19

The plant is at 100% power with a core flow of 69 Mlbm/hr when the following indications are received:

- Alarm Recirc Flow Converter Failure, C904R-A1, annunciates
- The output of Recirc Flow Converter "A" is observed to be zero

Assuming no manual action is taken, which one of the following is correct?

APRM "A" channel Hi-Hi setpoints have \_\_\_ (1) \_\_\_ (increased, decreased, remained the same).

The result of this event will be a \_\_\_ (2) \_\_\_.

- A. (1) decreased  
(2) ½ scram and rod block
- B. (1) decreased  
(2) rod block only
- C. (1) increased  
(2) ½ scram and rod block
- D. (1) remained the same  
(2) rod block only

Proposed Answer: A

Explanation (Optional):

- A. Correct: The "A" flow Converter has failed downscale. This results in a zero core flow input to all "A" side APRMs. This will cause the APRM scram and rod block setpoints to lower to those associated with a zero core flow. The scram setpoint for this condition is ~ 70% with the rod block setpoint at ~ 63%. With power at 100%, a ½ scram will occur.



A rod block will occur due to both an APRM Hi and a Flow Converter Downscale trip.

- B. Incorrect: A ½ scram will be generated. Plausible in that only a rod block would be generated if the flow converter had failed upscale.
- C. Incorrect: Scram setpoints would decrease. Plausible if the candidate is confused as to the response of the flow biased settings on the APRMs as this would be true if the flow converter had failed upscale. Additionally the candidate may think that Recirc flow converter failure may generate a ½ scram due to the flow converter going Inop.
- D. Incorrect: Scram setpoints would decrease. Plausible if the candidate focuses on the high core flow and the APRM scram clamp at 118% for any core flow above 55Mlbm/hr which prevents any increase in flow biased scram setpoints above 55Mlbm/hr. If so then only a rod block would occur due to the flow converter failure.

Technical Reference(s): Pilgrim Power to Flow Map (Attach if not previously provided)

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

Learning Objective: O-RO-02-07-04, EO 5d, e (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	
	K/A #	218000	A4.07
	Importance Rating	3.5	

Ability to manually operate and/or monitor in the control room: ADS valve acoustical monitor noise: Plant-Specific

Proposed Question: RO Question # 20

EOP-16, RPV Flooding is in progress.

- All SRVs have been opened
- Flooding is in progress using the condensate system

Which one of the following would indicate that the RPV has been flooded to the level of the Main Steam lines?

- RPV pressure lowers to 30 psig and stabilizes.
- Torus water level rises initially and then stabilizes.
- SRV tailpipe temperatures reach saturation temperature for the given RPV pressure.
- SRV acoustic monitor indications extinguish as RPV pressure lowers and then sporadically re-illuminate.

Proposed Answer: D

Explanation (Optional):

- Incorrect: This is an indication that the SRVs have closed. They are designed to stay physically open down to ~50 psig. A stable pressure would indicate that they have not yet been forced open by raising pressure and forcing them to re-open.
- Incorrect: Since a system external to the containment is being used, torus water level would continue to rise. Plausible in that this would be an indication if only systems aligned to the torus were being used.
- Incorrect: When the main steam lines are flooded and the SRVs are now passing water, the tail pipe temperatures would lower to subcooled values.
- Correct: As pressure lowers, the acoustic monitor lights would begin to extinguish as they are passing less flow. When the SRVs close at ~ 30 to 50 psig, they would

extinguish. Once level was raised and the pressure began to rise they would be forced back open. However PNPS 5.3.35 cautions that the acoustic monitors may respond erratically to subcooled flow.

Technical Reference(s): PNPS 5.3.35, section 4.3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 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 #	2	
	Group #	1	
	K/A #	262002	2.1.7
	Importance Rating	2.7	

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

Proposed Question: RO Question # 21

Given the following:

- A sustained loss of 120 VAC Vital Bus Y-2 occurred
- A manual reactor scram is inserted.

Following the scram the following indications are observed:

- The C905 Group Scram Logic Lights de-energized
- 135 Blue scram lights illuminated
- The following alarms have annunciated:
  - SPVAH Pressure Lo, C905R-F1
  - RPIS INOP C905L-D4,
  - RWM ROD BLOCK" C905L-A4
- After several minutes all IRMs indicate ~50 on Range 10 using back panel indications

Which of the following methods would be effective in inserting any withdrawn control rods?

Method 1: Venting the scram air header.

Method 2: Venting the over-piston area of control rods.

Method 3: Inserting control rods by using the Continuous Insert feature of the Rod Movement Control Switch

Method 4: Inserting control rods by using the Emergency In Feature of the Emergency In / Notch Override Switch

- A. Method 2 only
- B. Methods 1 and 3 only

- C. Methods 1 and 4 only
- D. Methods 2 and 4 only

Proposed Answer: A

Explanation (Optional):

- A. Correct – The only method that would insert a control rod for these indications is venting the over-piston area of the control rods. With the Group Scram Logic Lights de-energized and the Blue scram lights illuminated this indicates a hydraulic failure has occurred. Venting the air header would be ineffective as it is already vented. Methods 3 and 4 would be ineffective as with a loss of the vital bus Y2 there is no rod select power available.
- B. Incorrect: Method 1 would be ineffective as the air header is already vented. Method 3 is not available as with Y-2 de-energized there is no rod select power.
- C. Incorrect: Method 1 would be ineffective as the air header is already vented. Method 4 is not available as with Y-2 de-energized there is no rod select power.
- D. Incorrect: With a loss of the vital bus Y2 there is no rod select power available.

Technical Reference(s): 5.3.23, Sect 3.1, pg 6 (Attach if not previously provided)  
5.3.6, Sect. 5.0 [9] pg 6

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-04, EO-23 (As available)

Question Source: Bank # PNPS 3191  
Modified Bank #  
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 6

55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000	2.1.2
	Importance Rating	4.1	

Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation. (Component Cooling Water)

Proposed Question: RO Question # 22

A major winter storm has hit the plant site resulting in the total loss of intake structure equipment.

The plant has been scrammed and offsite power is still available. Plant parameters are stable and in their normal band following the scram.

Which one of the following is an approved method of maintaining drywell cooling that can be implemented after the scram IAW PNPS 5.3.3 "Loss of all Service Water"?

- A. Place RHR Loop 'A' in torus cooling and cross tie RBCCW Loops, "A" RBCCW Loop supplying.
- B. Place RHR Loop 'A' and RHR Loop "B' in torus cooling. RBCCW Loop 'A' will cool both loops of RHR.
- C. Establish public water supply to the fire water connection on the on the RBCCW supply to the RBCCW Loop "B" heat exchanger.
- D. Place RHR Loop 'A' in torus cooling and establish a Fire Water connection directly to the RBCCW side of the RHR Loop 'A' heat exchanger.

Proposed Answer: A

Explanation (Optional):

- A. Correct. Per PNPS 5.3.3 "Loss of all Service Water" this method has been approved as an immediate way to maintain drywell cooling using the Torus as heat sink for drywell heat loads
- B. Incorrect. Only one loop of RHR is placed in service and both loops of RBCCW are cross-tied.

- C. Incorrect. Fire water connection is on the SSW side of the heat exchanger.
- D. Incorrect. RHR is not needed for this lineup and the fire water connection is on the SSW side of the heat exchanger.

Technical Reference(s): PNPS 5.3.3, Section 5.2 [1] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-02-02, 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 10  
 55.43

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

Comments:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004	A2.05
	Importance Rating	3.3	

Ability to (a) predict the impacts of the following on the SOURCE RANGE MONITOR (SRM) SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Faulty or erratic operation of detectors/system

Proposed Question: RO Question # 23

A Reactor Startup is in progress with the following conditions:

- Reactor is critical on a stable, positive period.
- Reactor power is on range 3 of the IRMs.
- SRM detector "A" is stuck and cannot be withdrawn.
- SRM "A" indication rises to 5E5 cps.

Given the above, which one of the following is correct regarding,

(1) Any automatic actions

AND

(2) Any required actions directed by plant procedures that would allow the startup to continue?

- A. (1) SRM rod block and upscale alarm.  
(2) Bypass SRM "A"
- B. (1) SRM rod block and upscale alarm.  
(2) Place SRM "A" function switch to standby.
- C. (1) Upscale alarm only.  
(2) Bypass SRM "A".
- D. (1) Upscale alarm only.  
(2) Place SRM "A" function switch to standby.

Proposed Answer: A

Explanation (Optional):

- A. Correct: If any SRM channel detects > 1 X 105 cps, a rod block is activated. This rod block is bypassed if:
  - The mode switch is in RUN
  - The associated IRM range switches are at range 8 or above

Since the IRMs are on range 3, the rod block would occur.

IAW the ARP, the directed action is to bypass the SRM.

- B. Incorrect: Taking the function switch out of operate will cause an INOP trip and associated rod block. Plausible if the candidate does not remember that taking the function switch out of Operate will cause an INOP trip. Additionally this action is not directed by any plant procedure.
- C. Incorrect: A rod block also occurs. This answer would be correct if IRMs were on range 8 or higher. The candidate that confuses the range 8 IRM interlock with the range 3 interlock and does not understand the effect of the function switch may choose this answer. Additionally this action is not authorized by the procedure.
- D. Incorrect: A rod block would also occur. Additionally this action is not authorized by the procedure.

Technical Reference(s): PNPS 2.2.64, Section 4.3 (Attach if not previously provided)  
ARP C905L – E9

Proposed References to be provided to applicants during examination: None

Learning Objective: SOURCE RANGE MONITORING, EO (As available)  
6

Question Source: Bank # WTSI 4904  
Modified Bank #  
New

Question History: Last NRC Exam: Cooper 2008

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 #	1	
	K/A #	262001	A3.04
	Importance Rating	3.4	

Ability to monitor automatic operations of the A.C. ELECTRICAL DISTRIBUTION including:

Load sequencing

Proposed Question: RO Question # 24

A loss of offsite power has occurred coincident with a loss of coolant accident.

The "B" diesel generator has started and has re-energized 4160 VAC Bus A-6.

Which of the following represents the correct order for the sequencing of loads onto Bus A-6?

- A. RHR Pump "B" starts  
RHR Pump "D" starts  
Core Spray Pump "B" starts  
SSW Pump "D" starts  
RBCCW Pump "D" starts
- B. Core Spray Pump "B" starts  
RHR Pump "B" starts  
RHR Pump "D" starts  
SSW Pump "D" starts  
RBCCW Pump "D" starts
- C. RHR Pump "B" starts  
RHR Pump "D" starts  
Core Spray Pump "B" starts  
RBCCW Pump "D" starts  
SSW Pump "D" starts
- D. Core Spray Pump "B" starts  
RHR Pump "B" starts  
RHR Pump "D" starts  
RBCCW Pump "D" starts  
SSW Pump "D" starts

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The Core Spray pump starts first.
- B. Correct: The following sequence of events would automatically occur if a LOCA occurred simultaneously with a loss of off-site power.

Time Zero:

Diesel generator breakers close.  
Start of both Core Spray pumps

5 Seconds From Time Zero:

Start of first two RHR pumps (A and B).  
Core spray pumps at speed.

10 Seconds From Time Zero:

Start of next two RHR pumps (C and D).  
First two RHR pumps at speed.

20 Seconds From Time Zero:

SSW pump D starts (< 2.3 # w/a 20 second time delay).  
Vital AC MG set returns to AC power.

25 Seconds From Time Zero

SSW pump A starts if pressure is not normal (<2.3 # w/a 25 second time delay).

30 Seconds From Time Zero

RBCCW pumps A and D start (< 60 # w/a/ 30 second time delay)

- C. Incorrect: The Core Spray pump starts first. The SSW pump starts before the RBCCW pump
- D. Incorrect: The SSW pump starts before the RBCCW pump.

Technical Reference(s): PNPS 2.4.16, Attachment 7 (Attach if not previously provided)

Emergency AC Distribution  
reference text, pages 23 and 24

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-08 EO 11 (As available)

Question Source: Bank # WTSI 12470  
Modified Bank #

New

Question History: Last NRC Exam: 2010 Riverbend

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

10 CFR Part 55 Content: 55.41 7  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005	K5.06
	Importance Rating	2.5	

Knowledge of the operational implications of the following concepts as they apply to AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM : Assignment of LPRM's to specific APRM channels

Proposed Question: RO Question # 25

The plant is at power. APRM "A" is currently reading 100%. There are no bypassed LPRMs.

Then, an LPRM associated with an APRM "A" fails downscale. No operator action has yet been taken.

Following the failure what will the APRM Back Panel meter indicate if the Meter Function switch is placed in the COUNT position?

- A. 65
- B. 70
- C. 75
- D. 80

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: The meter will indicate 80 (see explanation for "D"). Plausible if the candidate confuses the number of LPRMs assigned to "A" side APRMs and "B" side APRMs and concludes that "A" APRM has only 14 LPRMs. Then if the candidate concludes that with the LPRM failing downscale there are only 13 remaining. However this would also be incorrect. Unlike the RBM which utilizes downscale trip units that will automatically remove a failed LPRM from its averaging circuit, this is not the case for the APRMs. If so this would result in an indication of 65 (13 X 5%=65).
- B. Incorrect: The meter will indicate 80. Plausible if the candidate confuses the number of LPRMs assigned to "A" side APRMs and "B" side APRMs and concludes that "A" APRM has only 14 LPRMs. Then if the candidate correctly concludes that the COUNT circuit counts all LPRMs that have not been bypassed will conclude that the meter will read 70% (14 X 5%=70%)

- C. Incorrect: The meter will indicate 80. Plausible if the candidate correctly identifies that there are 16 LPRMs assigned to "A" APRM but assumes that the LPRM failing downscale will automatically be removed from the COUNT Circuit and that only 15 remain ( $15 \times 5\% = 75$ )
- D. Correct: There are 16 LPRMs assigned to APRM "A". The COUNT circuit determines the number of assigned LPRMs which are in OPERATE (5 percent per LPRM). Since the LPRM has not yet been bypassed, the indication will remain at 80 ( $16 \times 5\% = 80$ ).

Technical Reference(s): APRM Reference text Page 7 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: N/A

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

10 CFR Part 55 Content: 55.41 7  
 55.43

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

Comments:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003	2.4.31
	Importance Rating	4.2	

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures. (IRM)  
Proposed Question: RO Question # 26

A reactor startup is in progress. All IRMs are at 50/125 on range 4.

Then, the following alarms are observed:

- IRM Hi alarm, C905L-C9
- Rod Withdrawal Block alarm, C905L-B4
- Auto Scram Chan "A" alarm, C905R-A1
- Neutron Monitoring Trip alarm, C905R-C3

Which one of the following inappropriate actions is consistent with these indications?

IRM E ...

- A. being down ranged to range 3.
- B. being up ranged to range 5.
- C. being withdrawn from the core.
- D. function switch taken out of Operate.

Proposed Answer: A

Explanation (Optional):

- A. Correct: Since the odd numbered ranges are just an expanded view of the even number ranges, the IRM would try to indicate 50 on range 3. Range 3 has a scale of 0-40. This would result in exceeding both the Hi and Hi-Hi trip setpoints which produce the indicated alarms
- B. Incorrect: This would result in downscale indications as the IRM would indicate 5/125 of scale on range 5.

- C. Incorrect: This would result in downscale indications.
- D. Incorrect: This would generate an INOP trip. An INOP trip would not result in the IRM Hi alarm.

Technical Reference(s): IRM Reference Text, pages 10 and 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-02, EO 14 (As available)

Question Source: Bank # WTSI 1262  
Modified Bank #  
New

Question History: Last NRC Exam: 2007 Fermi

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

10 CFR Part 55 Content: 55.41 7  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000	K1.14
	Importance Rating	2.5	

Knowledge of the physical connections and/or cause- effect relationships between REACTOR WATER CLEANUP SYSTEM and the following: Process sample system  
Proposed Question: RO Question # 27

The reactor is at rated conditions when the following occur:

- Steam Leakage Area Temp Hi, C904L-A6, alarms
- RWCU isolates
- The alarm is determined to be due to RWCU Heat Exchanger Room high area temperature

What action is now required regarding Recirc Loop Sample Valves AO-220-44 and AO-220-45 and why?

The valves should be .....

- A. Opened or verified open to aid in depressurizing the RWCU system.
- B. Closed or verified closed to prevent damage to the Mitigation Monitoring System.
- C. Closed or verified closed to prevent leakage from the reactor to the reactor building.
- D. Opened or verified open to re-establish a flow of water to the RWCU conductivity elements.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The valves are closed. Plausible since the indications provided are indicative of a RWCU leak. If the candidate remembers that there is an interface between the RWCU and the Mitigation Monitoring System (MMS) through the Recirc Loop Sample Valves the candidate may conclude that this is an appropriate action.
- B. Incorrect: Plausible if the candidate remembers that there is an interface between the RWCU and the Mitigation Monitoring System through the Recirc Loop Sample Valves the candidate may conclude that this is an appropriate action.

- C. Correct: Both the ARP for C904L-A6 and 2.4.47 direct that the Recirc Loop Sample valves be closed if RWCU isolates on high area temperature or flow. Isolation of the MMS is necessary if a valid steam leak detection signal has isolated the RWCU System. This is done by closing the Recirc Loop Sample valves. Valid steam leak detection signals are steam leakage area temp HI and/or inlet flow HI. This action prevents the MMS from providing a leakage pathway from the Reactor to the Secondary Containment.
- D. Incorrect: The valves are closed.

Technical Reference(s): ARP C904L-A6 (Attach if not previously provided)  
 PNPS 2.4.27, Subsequent Action  
 [1] and discussion item [3]

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-05, 20a (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 4  
 55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	226001	K1.08
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause-effect relationships between RHR/LPCI: CONTAINMENT SPRAY SYSTEM MODE and the following: Nuclear boiler instrumentation  
Proposed Question: RO Question # 28

A small break LOCA in the drywell has occurred.

Current plant conditions are as follows:

- 'A' Loop of RHR is has been placed in drywell spray mode IAW PNPS 5.3.35.2, Attachment 16, Drywell Spray Checklist
- Drywell pressure is currently 6 psig and slowly lowering.
- Reactor pressure is 420 psig and slowly lowering.
- RPV level is -40 inches and slowly lowering.

Without further operator action, RHR pump flow will remain aligned for drywell spray until:

- Reactor level lowers to -46 inches
- Reactor level lowers to -150 inches
- Drywell pressure lowers to 2.2 psig
- Reactor pressure lowers to 400 psig

Proposed Answer: B

Explanation (Optional):

- Incorrect: In order to establish drywell sprays initially the LPCI initiation signal had to be placed in override because of the high drywell pressure condition. RPV level lowering to -46 inches will have no additional impact.
- Correct: Although the LPCI initiation signal has been overridden, when RPV level lowers to the 2/3rds coverage interlock, -150 inches, the valves will close until the 2/3<sup>rd</sup> core coverage interlock is overridden.

- C. Incorrect: Although a lowering drywell pressure will eventually close the spray valves, this will not occur until drywell pressure lowers to 1.8 psig. Plausible in that 2.2 psig is the LPCI initiation setpoint.
- D. Incorrect: The drywell spray valves will remain open when pressure lowers to 400 psig. Plausible in that the LPCI system will realign for LPCI injection at this pressure. However since the LPCI initiation signal was already overridden the valves will remain open.

Technical Reference(s): RHR Reference Text, pages 44-46. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-01, EO 14 (As available)

Question Source: Bank # WTSI # 2172  
 Modified Bank #  
 New

Question History: Last NRC Exam: Pilgrim 2002

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

10 CFR Part 55 Content: 55.41 7  
 55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215002	K3.01
	Importance Rating	3.3	

Knowledge of the effect that a loss or malfunction of the ROD BLOCK MONITOR SYSTEM will have on following: Reactor manual control system: BWR-3,4,5  
Proposed Question: RO Question # 29

The plant is at 90% power with a center control rod selected.  
Then, the averaging circuit for RBM Channel "A" malfunctions such that the output of the RBM lowers to 84%.

Which one of the following is correct?

- A. No RBM trips will occur. No alarms or rod blocks will occur.
- B. A RBM INOP trip will occur generating a rod withdraw block and associated alarms.
- C. A RBM DOWNSCALE trip will occur generating a rod withdraw block and associated alarms.
- D. A RBM DOWNSCALE trip and associated alarm will occur. A rod withdraw block will not occur.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: A DOWNSCALE trip will occur. Plausible if the candidate realizes that the condition will not result in an INOP trip and confuses the downscale trip setting with that of all other instruments of ~ 5%.
- B. Incorrect: A DOWNSCALE trip will occur. Plausible if the candidate believes that an INOP trip will occur for this condition.
- C. Correct: A RBM downscale trip occurs at 94% and will result in a rod block, a downscale alarm and rod block alarm.

D. Incorrect: A rod block will occur.

Technical Reference(s): RBM Reference text, Figure 2. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-05, EO 2h (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 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	
	K/A #	201002	K4.07
	Importance Rating	2.5	

Knowledge of REACTOR MANUAL CONTROL SYSTEM design feature(s) and/or interlocks which provide for the following: Timing of rod insert and withdrawal cycles (rod movement sequence timer)

Proposed Question: RO Question # 30

During a single notch withdraw of a control rod, the RMCS timer applies a withdraw signal for 3.0 seconds.

Which one of the following is correct regarding future rod movement?

- A. The rod will settle into the next notch. Additional withdrawals are permissible provided they are within the withdraw limits of the RWM.
- B. A rod block will terminate the rod withdraw. Withdrawal of any other rod is also blocked. Insertion of any rod is allowed.
- C. The rod will be de-selected preventing further withdrawal. All other rods can be inserted or withdrawn as required.
- D. The rod will be de-selected preventing further withdraw. Rod selection of all remaining rods is blocked.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: A select block will be generated which will de-select the rod. Plausible if the candidate believes that 3 seconds is within the limits of the Aux timer.
- B. Incorrect: A select block is generated. Plausible if the candidate knows withdraw will be prevented but does not understand that a select block is generated.
- C. Incorrect: All other rods are prevented from being inserted or withdrawn due to the inability to select any rod.
- D. 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 (in panel 928) 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.

Technical Reference(s): RMCS Reference Text, page 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-08, EO 10a (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	
	K/A #	201001	K5.02
	Importance Rating	2.6	

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

Proposed Question: RO Question # 31

The plant is at rated conditions when the operator notes that the CRD flow indication on the CRD Flow controller is 75 gpm.

Which one of the following is a potential adverse consequence of this indication?

- A. Control rods drifting in.
- B. Control rods drifting out.
- C. Control rod accumulator Trouble Alarms.
- D. Control rod drive filter break through and clogging of the directional control valves.

Proposed Answer: A

Explanation (Optional):

- A. Correct: PNPS 2.4.11, Attachment 2 directs that if a rod is drifting in then check that the cooling water differential pressure is not too high. A CRD Flow pegged high would result in excessive cooling water differential pressure. Since cooling water flow is felt on the underside of the CRDM drive piston this could result in a rod drifting into the core.
- B. Incorrect: Rods will not drift out. The cooling water pressure is felt on the underside of the drive piston.
- C. Incorrect: An accumulator will alarm if pressure drops to 965 psig. Although pressure will drop in the CRD system it will not drop any lower than reactor pressure – otherwise flow would not increase. Reactor pressure at rated conditions is significantly higher than 965 psig.
- D. Incorrect. A high dp alarm would be received prior to this occurring. Additionally, there is a Y- Strainer downstream of the filter to prevent clogging of the directional control valves.

Technical Reference(s): PNPS 2.4.11, Attachment 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-11, EO 2g (As available)

Question Source: Bank #  
Modified Bank # PNPS Bank 2828 Modified stem so that FCV fails open not closed.  
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 2  
55.43

General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	214000	K6.02
	Importance Rating	2.7	

Knowledge of the effect that a loss or malfunction of the following will have on the ROD POSITION INFORMATION SYSTEM : Position indication probe

Proposed Question: RO Question # 32

While withdrawing a control rod from position 24 to 26, reed switch 25 fails to open. The rod is presently latched at notch 26. The expected indication(s) for this condition is/are:

- A. Black-black (no indication) for the rod on the four-rod display ONLY.
- B. Red drift light lit on the full core display, "25" displayed on the four-rod display.
- C. Red drift light lit on the full core display and black-black (no indication) for the rod on the four-rod display.
- D. Red drift light lit on the full core display, both "25" and "26" superimposed on one another on the four-rod display.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: This would be the indication if the position 26 reed switch failed to close. Plausible if the candidate does not understand whether reed switches are normally open or closed.
- B. Incorrect: "25" and "26" would be super imposed over one another.
- C. Incorrect: "25" and "26" would be super imposed over one another on the four-rod display.
- D. Correct: The drift circuit will annunciate if a control rod is NOT selected and driving and either its even numbered reed switch opens, or an odd numbered reed switch closes. Since the position "25" reed switch is stuck closed, a drift alarm will occur after the RMCS timer times out. Since two switches are closed, the two numbers will be superimposed over one another.

Technical Reference(s): RPIS Reference Text, page 8 and (Attach if not previously provided)  
13

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-04, EO-7 (As available)

Question Source: Bank # 4927  
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 2  
55.43

General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202001	A1.06
	Importance Rating	2.5	

Ability to predict and/or monitor changes in parameters associated with operating the RECIRCULATION SYSTEM controls including: Recirculation pump motor amps

Proposed Question: RO Question # 33

Given the following:

- The plant is at rated power
- "B" Recirc MG Set AC Lube Oil Pump P225B is running
- "B" Recirc MG Set AC Lube Oil Pump P225A is danger tagged out of service

Then...

- 125 VDC bus D-5 is lost
- Two minutes later the reactor operator reports that current on the "B" Recirc MG set motor is pegged high and that P225B is not running

Which one of the following actions is required by PNPS 5.3.12, LOSS OF ESSENTIAL DC BUS D17 OR D5 AND D37?

- Manually scram the reactor.
- Manually trip 4KV bus A4 at panel C3.
- Locally trip the "B" Recirc MG Drive motor breaker at 4KV bus A4.
- Locally start the "B" Recirc System DC Emergency Bearing Oil Pump at 250 VDC panel D9.

Proposed Answer: A

Explanation (Optional):

- Correct – Per the procedure, if there is indication of severe bearing damage (locked rotor condition), then a Scram is initiated and the resulting Turbine trip will de-energize the Unit Auxiliary Transformer, which in turn results in de-energizing Bus A4 and thereby securing the "B" Recirc MG Set.

- B. Incorrect: P225B tripped as expected following the loss of D5. With no AC lube oil pumps running the MG set is seizing. However with the loss of D5, bus A4 is also without control power.
- C. Incorrect – The procedure directs that the MG set be tripped locally but only if the current is NOT pegged high due to the electrical hazard of tripping the breaker under such a high load.
- D. Incorrect: Starting the DC pump will not alleviate the condition as it only supplies oil to the coupler bearings and not the motor and generator bearings.

Technical Reference(s): PNPS 5.3.12 LOSS OF ESSENTIAL DC BUS D17 OR D5 AND D37, discussion section, item [3] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: 2011 Pilgrim # 56

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

10 CFR Part 55 Content: 55.41 10  
 55.43

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

Comments:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	286000	A2.12
	Importance Rating	3.1	

Ability to (a) predict the impacts of the following on the FIRE PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Low diesel fuel supply:

Proposed Question: RO Question # 34

The plant is experiencing station blackout conditions. The Diesel Driven Fire Pump is being used for vessel makeup.

Assuming that the Diesel Driven Fire Pump is being run at full capacity,

- (1) How long will the diesel continue to run without any operator action before it shuts down due to lack of fuel?

AND

- (2) What action can be taken to extend the time available?

- A.
  - (1) 8 hours
  - (2) Manually align the diesel oil transfer pump (P-141A) and transfer diesel fuel from the EDG fuel oil tanks to the diesel fire pump.
- B.
  - (1) 8 hours
  - (2) Manually align the hydro-turbine fuel oil transfer pump (P-181) and transfer diesel fuel from the EDG fuel oil tanks to the diesel fire pump.
- C.
  - (1) 24 hours
  - (2) Manually align the diesel oil transfer pump (P-141A) and transfer diesel fuel from the EDG fuel oil tanks to the diesel fire pump.
- D.
  - (1) 24 hours
  - (2) Manually align the hydro-turbine fuel oil transfer pump (P-181) and transfer diesel fuel from the EDG fuel oil tanks to the diesel fire pump.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The hydro-turbine fuel oil transfer pump (P-181) must be used. Plausible in that P-141A is normally used to transfer fuel oil from the EDG storage tanks to the Diesel Fire Pump day tank. However this pump is not DC powered and would not be available during blackout conditions.
- B. Correct: The diesel fire pump day tank is manually kept full from the EDG oil transfer system. A full day tank provides continuous operation for 8 hours. During loss of all AC, the fuel oil transfer system will be configured to operate the hydro-turbine fuel oil transfer pump per PNPS 2.4.54. The hydro-turbine pump does not require any power to operate. It uses the discharge from the fire pump to run a turbine which operates the transfer pump. The hydro-turbine fuel oil transfer pump is not for fire protection, but for "beyond design basis accidents" when the fire pump is used for Rx vessel makeup.
- C. Incorrect: A full day tank provides continuous operation for 8 hours. Plausible in that it is a "day" tank. Also The hydro-turbine fuel oil transfer pump (P-181) must be used.
- D. Incorrect: A full day tank provides continuous operation for 8 hours. However this pump is not DC powered and would not be available during blackout conditions.

Technical Reference(s): Fire Water Protection System Ref (Attach if not previously provided) text pages 15, 16 and 39

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-10-05, 5d, 9 (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

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 #	2	
	Group #	2	
	K/A #	288000	A3.01
	Importance Rating	3.8	

Ability to monitor automatic operations of the PLANT VENTILATION SYSTEMS including:  
Isolation/initiation signals

Proposed Question: RO Question # 35

The plant is refueling. An air purge of the drywell through the Contaminated Exhaust Fans is in progress.

Then, the following process rad monitor Upscale trips occur:

- Channel "A" Refuel Floor Exhaust RM 1705-8A
- Channel "B" Refuel Floor Exhaust RM 1705-8B

Given that there are NO other tripped PRMs which one of the following is correct regarding any automatic actions that may or may not occur?

- A. NO isolations occur
- B. Complete Reactor Building Ventilation isolation  
Drywell purge lineup does NOT isolate
- C. Complete Reactor Building Ventilation isolation  
Complete Drywell purge lineup isolation
- D. One of the two in-series Reactor Building Isolation Dampers isolate  
One of the two in-series Drywell Purge supply and exhaust valves isolate

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: Both the Reactor Building and the Purge lineup isolated. Plausible if the candidate does not understand the configuration of the PRMs in the RBIS logic and concludes that the RBIS logic does not actuate as there are some combinations of PRM trips that will not cause an isolation.
- B. Incorrect: The drywell purge lineup isolates. Plausible if the candidate does not realize

that an RBIS trip also isolates the Primary Containment Atmosphere Control Valves (PCAC).

- C. Correct: Refuel Floor Exhaust Channels A & B tripping will result in both RBIS logic train actuations this will cause a Reactor Building Ventilation isolation and a PCAC isolation.
- D. Incorrect: All Reactor Building and Drywell purge dampers will isolate.

Technical Reference(s): PCAC reference text page 24 & (Attach if not previously provided)  
Figures 9

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-05, EO 10, 11 (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 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	
	K/A #	241000	A4.11
	Importance Rating	3.1	

Ability to manually operate and/or monitor in the control room: Turbine speed (Reactor/Turbine Pressure Regulator)

Proposed Question: RO Question # 36

The Main Turbine is currently on the jack and is about to be rolled during a plant startup.

The Turbine is rolled off the jack and the initial Turbine acceleration is controlled using the \_\_\_ (1) \_\_\_.

While the Turbine speed is increasing, Reactor Pressure is maintained by the \_\_\_ (2) \_\_\_.

- A. (1) Load Limit Control Switch.  
(2) Bypass Valves
- B. (1) Speed/Load Changer Control Switch  
(2) Turbine Control Valves
- C. (1) Load Limit Control Switch.  
(2) Turbine Control Valves
- D. (1) Speed/Load Changer Control Switch  
(2) Bypass Valves

Proposed Answer: A

Explanation (Optional):

- A. Correct: The turbine is initially rolled using the load limit which is initially set to zero. As the control valves open slightly to accelerate the turbine the bypass valves close down to control pressure.
- B. Incorrect: The turbine is rolled with the Load Limit and pressure is controlled by the bypass valves. Both are plausible. The Speed load changer will limit the speed of the turbine to ~ 1710 RPM and is used to complete the acceleration of the turbine to rated speed. Since the control valves are opening and the control valves normally throttle to control pressure a candidate may assume that this is also true during the initial roll.

- C. Incorrect: The bypass valves will throttle as required to control pressure.
- D. Incorrect: The load limit is used initially to roll the turbine.

Technical Reference(s): PNPS 2.1.1, steps [111] thru [115] (Attach if not previously provided) and [127]

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-05-04, 13e, f, 14a (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  
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	
	K/A #	219000	2.1.32
	Importance Rating	3.8	

Conduct of Operations: Ability to explain and apply all system limits and precautions (RHR/LPCI: Torus/Pool Cooling Mode)

Proposed Question: RO Question # 37

The plant is at rated conditions.

“A” Loop of RHR is placed in Torus Cooling in preparation for HPCI Testing.

IAW 2.2.19, Residual heat Removal, which one of the following Tech Spec actions is required prior to placing “A” RHR into Torus Cooling and why?

- A. Declare Primary Containment inoperable. If a loss of offsite power was to occur and the “A” EDG failed to start, the Torus Cooling valves will be inoperable in the open position.
- B. Declare LPCI inoperable. If a loss of offsite power was to occur and the “A” EDG failed to start coincident with a LOCA, the Torus Cooling valves will lose power and remain open, diverting injection flow to the Torus.
- C. Declare LPCI inoperable. If a loss of offsite power was to occur while aligned for Torus Cooling the RHR loop will drain and create a potential for subsequent water hammer if an RHR pump was started.
- D. Declare “A” Loop of Torus Cooling Inoperable. If a loss of offsite power was to occur and the “A” EDG failed to start, the Torus Cooling valves will be inoperable in the open position.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: LPCI mode is declared inoperable. Plausible in that the situation described would be true.
- B. Correct: IAW of PNPS 2.2.19, section 4.1.2 item [3], the requirement to enter an Active LCO for the LPCI System being inoperable (Tech Spec 3.5.A) whenever the LPCI System is in the Torus Cooling mode of operation is based upon a postulated accident scenario that involves a LOOP-LOCA while in Torus Cooling and a single failure of an

EDG. With the loss of the EDG, the Torus Cooling valves will lose power and remain open, diverting flow to the Torus during the subsequent LPCI injection with the remaining two LPCI pumps.

- C. Incorrect: Plausible in that this would occur but is not the reason for declaring the system inoperable because the system is designed so that significant damage would not occur due to water hammer. TS Bases, page B3/4.5-24
- D. Incorrect: LPCI is declared inoperable.

Technical Reference(s): PNPS 2.2.19, section 4.1.2 item (Attach if not previously provided)  
[3].

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-01, EO 26 (As available)

Question Source: Bank # WTS 11830  
Modified Bank # (Note changes or attach parent)  
New

Question History: Last NRC Exam: 2010 NMP2

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 #	2	
	Group #	2	
	K/A #	259001	K4.04
	Importance Rating	2.5	

Knowledge of REACTOR FEEDWATER SYSTEM design feature(s) and/or interlocks which provide for the following: Dispersal of feedwater in the reactor vessel

Proposed Question: RO Question # 38

The reactor is at rated conditions. The operator notes the following feedflow indications:

- “A” Feedwater flow, FI-640-24A indicates 4.25 Mlbm/hr
- “B” Feedwater flow, FI-640-24B indicates 3.75 Mlbm/hr

How is the feedwater dispersed in the reactor vessel to ensure proper mixing of the feedwater and downcomer flow?

AND

Given the indications above how can this feedwater flow deviation be eliminated?

- The two feed lines downstream of the feed reg valves split into 2 lines so there are 4 vessel penetrations equally spaced around the vessel. The feedwater flows directly into the annulus area mixing with the downcomer flow.  
Equalize flow by adjusting the bias settings on the FWLC M/A Stations.
- The two feed lines downstream of the feed reg valves split into 2 lines so there are 4 vessel penetrations equally spaced around the vessel. The feedwater flows directly into the annulus area mixing with the downcomer flow.  
Equalize flow by placing one FWLC M/A Station in manual and adjusting as required.
- The two feed lines downstream of the feed reg valves split into 2 lines so there are 4 vessel penetrations equally spaced around the vessel. The feedwater then flows through 4 spargers to distribute the flow in the annulus area.  
Equalize flow by adjusting the bias settings on the FWLC M/A Stations.
- The two feed lines downstream of the feed reg valves split into 2 lines so there are 4 vessel penetrations equally spaced around the vessel. The feedwater then flows through 4 spargers to distribute the flow in the annulus area.  
Equalize flow by placing one FWLC M/A Station in manual and adjusting as required.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The feedflow is distributed to the downcomer annulus via feedwater spargers.
- B. Incorrect: The feedflow is distributed to the downcomer annulus via feedwater spargers. Additionally, PNPS 2.2.82 directs using the bias settings to equalize flow.
- C. Correct: Four 12-inch nozzles direct feedwater into the reactor vessel to replace the steam sent to the turbine. The feedwater nozzles discharge through the feedwater spargers to the downcomer annulus. PNPS 2.2.82 directs using the bias settings to equalize flow.
- D. Incorrect: PNPS 2.2.82 directs using the bias settings to equalize flow.

Technical Reference(s): PNPS 2.2.82 Precaution 5.0 (Attach if not previously provided)

Condensate and Feed Reference  
Text page 30.  
Reactor Vessel and Internals  
Reference Text, page 16

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-07, EO 5a (As available)  
O-RO-02-04-10, EO 13

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 X  
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 2  
55.43

General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023	AK1.01
	Importance Rating	3.6	

Knowledge of the operational implications of the following concepts as they apply to  
**REFUELING ACCIDENTS : Radiation exposure hazards**

Proposed Question: RO Question # 39

Core Alterations are in progress when the following occurs:

- An irradiated fuel bundle is being moved from the reactor cavity to the Spent Fuel Pool
- The Bundle becomes ungrappled and falls into the reactor vessel downcomer area.  
(Between the vessel wall and the shroud)
- Bundle integrity is maintained

Which one of the following workers is at greatest risk of radiation overexposure?

- A. I&C Tech at the SBLC tank
- B. Refuel SRO on the Bridge
- C. Mechanic working on SRVs
- D. RP Tech at Refuel Floor Access Point

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: because of the location of these components. SRO on the bridge is shielded by water level within the cavity, as is the RP Tech at the access. SLC Tank is in Secondary Containment, which is shielded by Primary Containment wall.
- B. Incorrect: because of the location of these components. SRO on the bridge is shielded by water level within the cavity, as is the RP Tech at the access. SLC Tank is in Secondary Containment, which is shielded by Primary Containment wall.
- C. Correct: because of the location of these components. SRO on the bridge is shielded by water level within the cavity, as is the RP Tech at the access. SLC Tank is in Secondary Containment, which is shielded by Primary Containment wall.

D. Incorrect: because of the location of these components. SRO on the bridge is shielded by water level within the cavity, as is the RP Tech at the access. SLC Tank is in Secondary Containment, which is shielded by Primary Containment wall.

Technical Reference(s): Plant layout drawing (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS Bank 1048  
Modified Bank #  
New

Question History: Last NRC Exam: 2008 Nine Mile Point 2

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 #	1	
	Group #	1	
	K/A #	295028	EK1.02
	Importance Rating	2.9	

Knowledge of the operational implications of the following concepts as they apply to HIGH DRYWELL TEMPERATURE : Equipment environmental qualification

Proposed Question: RO Question # 40

The drywell has a maximum internal design temperature of \_\_ (1) \_\_ and the ADS SRV solenoids are designed to operate up to an ambient drywell temperature of \_\_ (2) \_\_.

	(1) <u>Maximum Drywell Internal Temperature</u>	(2) <u>Maximum SRV Solenoid Temperature</u>
A.	215°F	281°F
B.	215°F	330°F
C.	281°F	281°F
D.	281°F	330°F

Proposed Answer: D

Explanation (Optional):

- A. Incorrect : 215°F is the limiting drywell temperature to guarantee that ECCS trips occur on/or before present T.S. values and requires a shutdown be initiated if drywell temperature cannot be restored to less than 215°F within 30 minutes. 281°F is the maximum internal design temperature of the drywell.
- B. Incorrect : 215°F is the limiting drywell temperature to guarantee that ECCS trips occur on/or before present T.S. values and requires a shutdown be initiated if drywell temperature is restored less than 215°F within 30 minutes.
- C. Incorrect: The ADS SRV solenoids are designed to operate up to an ambient drywell temperature of 330°F.
- D. Correct: IAW Primary Containment reference text Section C.1. , the maximum internal design temperature of the drywell is 281 degrees F. The ADS SRV solenoids are designed to operate up to an ambient drywell temperature of 330°F. Main Steam

System reference text page 16 - Engineering determined that the ambient temperature inside the drywell during a main steam line break may rise to 330°F The design basis MSLB is the most limiting break. This would cause the nitrogen inside the accumulator to increase in pressure due to the lack of area for expansion. The pressure would continue to rise to 160 psi. The 160 psi pressure exceeds the design rating of the previous accumulator (135 psi) solenoid valves. For this reason, new solenoid valves and relief valves designed to operate with a maximum Nitrogen pressure of 160 psid were installed on the accumulators. The maximum internal design temperature of the drywell is 281°F.

Technical Reference(s): Main Steam System reference text (Attach if not previously provided)  
page 14  
Primary Containment Structure  
reference text page 8

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-04-01, EO6 (As available)  
O-RO-03-04-05, EO11

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

Question History: Last NRC Exam: Pilgrim 2011, # 44

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

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	
	K/A #	295004	AK1.04
	Importance Rating	2.8	

AK1.04 - Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER : Effect of battery discharge rate on capacity

Proposed Question: RO Question # 41

Station blackout conditions have existed for the past 5 hours.

Which of the following actions specified in PNPS 5.3.31, Station Blackout, will reduce the discharge rate on the "A" 125 VDC battery and extend the battery's capacity?

- Action 1: Stopping Recirc MG set "A" DC Emergency Bearing Oil Pump
- Action 2: Stopping the Turbine Emergency Bearing Oil Pump
- Action 3: Transferring 125VDC bus D-6 to its alternate supply

- A. Action 1 only
- B. Action 3 only
- C. Actions 1 and 2 only
- D. Actions 2 and 3 only

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Action 1 is associated with a 250 VDC load. Plausible in that this is an action addressed in PNPS 5.3.31 for reducing overall loads on the DC distribution system.
- B. Correct: IAW subsequent actions of PNPS 5.3.31, step [3] (e), if power has NOT been restored to MCC B15, THEN, AFTER approximately 5 hours, TRANSFER the 125V DC "swing bus" (D6) from the "A" 125V battery to the "B" 125V battery.

- C. Incorrect: Action 2 is associated with a 250 VDC load. Plausible in that this is an action addressed in PNPS 5.3.31 for reducing overall loads on the DC distribution system.
- D. Incorrect: Actions 2 and 3 are associated with 250 VDC loads. Plausible in that these actions are addressed in PNPS 5.3.31 for reducing overall loads on the DC distribution system.

Technical Reference(s): PNPS 5.3.31, step [3] (e) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: 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 #	1	
	Group #	1	
	K/A #	600000	AK2.01
	Importance Rating	2.6	

Knowledge of the interrelations between PLANT FIRE ON SITE and the following: Sensors, detectors and valves

Proposed Question: RO Question # 42

The plant is at 100% when a Main Transformer Lockout results in a scram. A report of fire 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 actuate 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

Explanation (Optional):

- 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 Transformer lockout when "rate of rise" temperature indicators trip
- C. Correct – Deluge system actuates from following a Main Transformer 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): 5.5.2, Att. 18, pg 82, 85 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-10-05, Terminal Objective, (As available)  
MANUALLY ACTIVATE FIRE  
PROTECTION SYSTEM.

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 4  
55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019	AK2.09
	Importance Rating	3.3	

Knowledge of the interrelations between PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR and the following: Containment

Proposed Question: RO Question # 43

The plant is at rated conditions with all systems aligned for normal full power operation.

Then, annunciator AIR/N2 TO DRYWELL TROUBLE, C904LC F-3, alarms.

Pneumatic supply pressure to the drywell on panel C7 is 102 psig and stable.

This means that Drywell pneumatics has automatically swapped to the ....

- A. nitrogen storage tank
- B. backup nitrogen bottles
- C. instrument air system supply
- D. SRV backup nitrogen accumulator supply

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: At 100% power Nitrogen is aligned to drywell pneumatics. The Nitrogen Storage tank is the normal source of nitrogen and maintains the supply header at ~ 110 psig. This alarm means that pressure from the Nitrogen Storage tank has lowered and the Backup nitrogen bottles are now in service.
- B. Correct: At 100% power Nitrogen is aligned to drywell pneumatics. The Nitrogen Storage tank is the normal source of nitrogen and maintains the supply header at ~ 110 psig. The backup nitrogen system is comprised of two sets of ten nitrogen cylinders. One bank is set to maintain the backup supply header at approximately 100 psig. The other bank will supply the header when the pressure drops to approximately 94 psig. If the instrument supply header pressure drops to 102 psig, annunciator "AIR/N2 TO DRYWELL TROUBLE" (C904LC-F3) will alarm and the instrument supply header will be

supplied by the backup nitrogen system. Instrument supply header pressure is indicated by PI-4348, on Panel C-7

- C. Incorrect: At 100% power, the drywell is inerted. When inerted, instrument air is valved out of service and manual action would be required to place it back in service.
- D. Incorrect: These indications mean that pneumatic supply has swapped to the backup N2 bottles. Plausible in that there is a backup supply for extended SRV control that must be valves in.

Technical Reference(s): PNPS 2.2.105 page 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-02, EO-3.0 (As available)

Question Source: Bank # 6049 Minor editorial changes  
Modified Bank #  
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 8  
55.43

Components, capacity, and functions of emergency systems.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025	EK2.10
	Importance Rating	2.9	

Knowledge of the interrelations between HIGH REACTOR PRESSURE and the following:  
 SPDS/ERIS/CRIDS/GDS: Plant-Specific  
 Proposed Question: RO Question # 44

Regarding the SPDS system status windows shown below:

LPCI	WTR AVAILABLE	RPV PR HIGH	PWR AVAILABLE	PMP OFF
SHTDN COOLING	CLG NOT AVAIL	RPV PR HIGH	PWR AVAILABLE	PMP OFF

This indication means that ...

- A. Torus water level is above the Tech Spec minimum.
- B. RPV pressure is above the shutoff head of the LPCI pumps.
- C. Cooling water is not available to the RHR pumps seal coolers.
- D. RPV pressure is high outside the allowable validation range and is suspect.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: "WTR AVAILABLE" means that Torus water level is above the minimum required for pump operation (~ 35 inches).
- B. Correct: IAW PNPS 2.6.1, page 65, RPV PR HI means that the RPV pressure is higher than the shutoff head pressure of the system pump.
- C. Incorrect: This means that pressure is above the shutoff head of the pump. Plausible in that the SDC Status is displaying "CLG NOT AVAIL". However this means that there is

no Cooling Water flow to the RHR HX or the RPV temperature is not higher than the Cooling Water inlet temperature to the RHR HXs (see page 69 of PNPS 2.6.1)

D. Incorrect: This means that pressure is above the shutoff head of the pump. Plausible in that a validation check is performed on status windows based on the status of sensors.

Technical Reference(s): PNPS 2.6.1, page 65, (Attach if not previously provided)  
DETERMINE PLANT SYSTEM  
STATUS USING EPIC-SPDS.

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-11-01, Terminal Objective, (As available)  
DETERMINE PLANT SYSTEM  
STATUS USING EPIC-SPDS.

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	
	K/A #	295026	EK3.04
	Importance Rating	3.7	

EK3.04 - Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: SBLC injection

Proposed Question: RO Question # 45

During ATWS conditions, reactor power is 10% following the tripping of the Recirc Pumps.

Given the above, what are the bases for injecting Boron before torus water temperature exceeds the Boron Injection Initiation Temperature?

- A. It ensures that the Cold Shutdown Boron Weight will be injected before the Torus temperature reaches the Heat Capacity Temperature Limit.
- B. It ensures that the Hot Shutdown Boron Weight will be injected before the Torus temperature reaches the Heat Capacity Temperature Limit.
- C. It ensures that the Cold Shutdown Boron Weight will be injected before the Torus bottom pressure reaches the Primary Containment Pressure Limit.
- D. It ensures that the Hot Shutdown Boron Weight will be injected before the Torus bottom pressure reaches the Primary Containment Pressure Limit.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The curve is based on injecting the Hot Shutdown Boron weight.
- B. Correct: THE BIIT curve is defined as the greater of either:
  - 1) The torus water temperature at which a reactor scram is required by plant Technical Specifications (110°F) or
  - 2) The highest torus water temperature at which initiation of boron injection using SBLC will result in injection of the Hot Shutdown Boron Weight (HSBW) before torus water temperature exceeds the Heat Capacity Temperature Limit (HCTL).

For reactor power levels below 12.5%, if boron is injected before torus water

temperature reaches BIIT, the torus temperature will remain below the HCTL during RPV blowdown. This ensures that all of the steam will be suppressed in the torus, and will minimize the pressure increase of the containment during a blowdown.

- C. Incorrect: The limit of concern is the HCTL, not the PCPL. Plausible in that one of the objectives of EOP-02 is to protect the primary containment by lowering water level when required. Additionally the curve is based on injecting the Hot Shutdown Boron weight.
- D. Incorrect: The limit of concern is the HCTL, not the PCPL. Plausible in that one of the objectives of EOP-02 is to protect the primary containment by lowering water level when required.

Technical Reference(s): O-RO-03-04-02, Page 77 and 78 (Attach if not previously provided)  
O-RO-03-04-04, Page 61

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-04, EO 20 (As available)

Question Source: Bank # WTSI 295026  
Modified Bank #  
New

Question History: Last NRC Exam: 2009 Hope Creek

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	
	K/A #	295021	AK3.01
	Importance Rating	3.3	

Knowledge of the reasons for the following responses as they apply to LOSS OF SHUTDOWN COOLING : Raising reactor water level

Proposed Question: RO Question # 46

Given the following conditions:

- The plant is depressurized with coolant temperature at 150 degrees F.
- MO-1001-50 Valve (Shutdown Cooling Suction) fails closed and cannot be opened by any means.
- Both Reactor Recirc Pumps are tagged out.

Under these conditions, PNPS 2.4.25, "Loss of Shutdown Cooling" requires reactor water level be \_\_\_\_\_.

- maintained below the Group I isolation setpoint to ensure that Bypass Valves are available for use in the event that the plant becomes pressurized.
- maintained below the HPCI Hi Level trip point to ensure that HPCI is available for use in the event that the plant becomes pressurized.
- raised above +60 inches to promote natural circulation.
- raised above +60 inches in preparation for initiating cooling by feed and bleed.

Proposed Answer: C

Explanation (Optional):

- Incorrect – must be maintained > +60 inches
- Incorrect – must be maintained > +60 inches
- Correct – Per PNPS 2.4.25, Loss of Shutdown Cooling if forced the reactor is NOT pressurized or if no heat sink is available, you must raise level above +60 inches in order to promote natural circulation or start a recirc pump to promote natural circulation.

D. Incorrect – feed and bleed is performed with the reactor pressurized

Technical Reference(s): PNPS 2.4.25 Step 4.0[6]b. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-01-18, TO DIRECT CORRECTIVE ACTIONS TO MITIGATE CONSEQUENCES OF ABNORMAL EVENT. (As available)

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

Question History: Last NRC Exam: 20011 Pilgrim, # 55

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

10 CFR Part 55 Content: 55.41 2, 10  
55.43

General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024	EK3.05
	Importance Rating	3.5	

EK3.05 - Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE : RPV flooding  
Proposed Question: RO Question # 47

A leak in the drywell has resulted in high drywell pressure and temperature. Emergency RPV Depressurization has been conducted in response to the containment parameters. Current plant conditions are as follows:

- Drywell pressure: 40 psig rising
- Torus Bottom pressure: 38 psig rising
- Drywell temperature: 340 °F, rising
- Reactor Pressure: 60 psig, lowering slowly
- All control rods are inserted
- Reactor level:
  - oscillating full scale on all Narrow Range instruments
  - oscillating between the bottom of scale and -80 inches on the Fuel Zone instruments
  - Shutdown and FWLC instruments are pegged high.

Which one of the following is required and why?

- A. Primary Containment Flooding in order to prevent containment failure.
- B. RPV Flooding in order to establish and maintain adequate core cooling via steam cooling.
- C. RPV Flooding in order to establish and maintain adequate core cooling via core submergence.
- D. Primary Containment Flooding in order to establish and maintain adequate core cooling via core submergence.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: RPV Flooding is required because RPV level cannot be determined. Plausible due to the very high containment parameters and that the RPV has already been depressurized.
- B. Incorrect: RPV Flooding floods the RPV to the main steam lines and establishes core cooling via core submergence.
- C. Correct: The high drywell temperature and low RPV pressure have resulted in plant conditions exceeding the RPV Saturation Temperature limit which may result in instrument run flashing. The indications provided are indicative that flashing is occurring and RPV level cannot be determined. RPV flooding is then required. RPV Flooding floods the RPV to the main steam lines and establishes core cooling via core submergence.
- D. Incorrect: RPV Flooding is required. Plausible in that Primary Containment Flooding is required if RPV water level cannot be restored and maintained above -150 inches and the Fuel Zones are oscillating below -150 inches.

Technical Reference(s): O -RO-03-04-08, page 7 (Attach if not previously provided)  
EOP Caution 1  
EOP-01, RPV Control, Overrides  
L-2 and P-1

Proposed References to be provided to applicants during examination: None

Learning Objective: O -RO-03-04-08, EO 1 and 3 (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 5  
55.43

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

changes, and operating limitations and reasons for these operating characteristics.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037	EA1.07
	Importance Rating	3.9	

Ability to operate and/or monitor the following as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN : RMCS: Plant-Specific

Proposed Question: RO Question # 48

PNPS 5.3.23, Alternate Rod Insertion, is being used to manually insert control rods during ATWS conditions.

The Rod Worth Minimizer has just been bypassed and drive water pressure has been increased to 400 psid.

IAW 5.3.23, which one of the following correctly describes how control rods are now inserted?

First insert ...

- A. the RPR array using the Emergency In / Notch Override Switch.
- B. the RPR array using the continuous insert feature of the Rod Movement Control Switch.
- C. control rods to form a checkerboard in a spiral pattern from the outside to the center using the Emergency In / Notch override Switch in the EMER IN position.
- D. control rods to form a checkerboard in a spiral pattern from the outside to the center using the continuous insert feature of the Rod Movement Control Switch.

Proposed Answer: A

Explanation (Optional):

- A. Correct: The RPR array is inserted first using the Emergency In Switch.
- B. Incorrect: 5.3.23 specifies that the Emergency In switch be used. Plausible in that placing and holding the Rod Movement Control Switch will cause the rod to continually insert (vice inserting just one notch).
- C. Incorrect: The RPR array is inserted first. Plausible in that after the RPR array is inserted then the rods are inserted to form a checkerboard pattern.

D. Incorrect: The RPR array is inserted first. Plausible in that after the RPR array is inserted then the rods are inserted to form a checkerboard pattern. Additionally, use of the Emergency In switch is specified.

Technical Reference(s): PNPS 5.3.23, Section 3.1, step [3](e) (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-04, TO IMPLEMENT REQUIRED CORRECTIVE ACTIONS FOR AN ATWS (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 X  
Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016	AA1.01
	Importance Rating	3.8	

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

ABANDONMENT : RPS

Proposed Question: RO Question # 49

The Main Control Room has been evacuated due to a fire. No operator actions were taken before the control room was evacuated.

IAW PNPS 2.4.143, Shutdown From Outside the Control Room

(1) What method is used to insert a reactor scram

AND

(2) Assuming that EPIC is not available what method is used to determine reactor shutdown status?

- A. (1) Manually venting the scram air header locally  
(2) Verifying the scram air header pressure has decayed to zero
- B. (1) Manually venting the scram air header locally  
(2) Verifying that the scram inlet and outlet valves have opened on all HCUs
- C. (1) Opening the Reactor Protection System Logic Channel "A" and "B" breakers on Panel C511  
(2) Verifying the scram air header pressure has decayed to zero
- D. (1) Opening the Reactor Protection System Logic Channel "A" and "B" breakers on Panel C511  
(2) Verifying that the scram inlet and outlet valves have opened on all HCUs

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: The method prescribed in PNPS 2.4.143 is to open the breakers on C511. Additionally since EPIC is not available the only method remaining to determine



shutdown status is by verifying the status of the scram inlet and out let valves on the HCU's.

- B. Incorrect: The method prescribed in PNPS 2.4.143 is to open the breakers on C511.
- C. Incorrect: Since EPIC is not available the only method remaining to determine shutdown status is by verifying the status of the scram inlet and out let valves on the HCU's. Plausible in that venting the air header is a method prescribed in PNPS 5.3.23 during ATWS conditions.
- D. Correct: PNPS 2.4.143 directs that if possible, the reactor be scrammed before the control room is evacuated. If not possible then the procedure via step [16](a) directs that Reactor Protection System Logic Channel "A" and "B" breakers on Panel C511 be opened which will de-energize RPS and should result in control rod insertion.

Step [16](g) then directs that the shutdown status be determined. With EPIC not available the only method available listed in Appendix M and as discussed in Discussion Step 5.0[9] is verifying that scram inlet and outlet valves have opened.

Technical Reference(s): PNPS 2.4.143, pages 10 and 12 (Attach if not previously provided)  
PNPS 2.4.143 Discussion Step  
5.0[9] and Appendix M

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #  
Modified Bank # PNPS 12691 Modified to include method for  
determining shutdown status.  
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 #	1	
	Group #	1	
	K/A #	295003	AA1.04
	Importance Rating	3.6	

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : D.C. electrical distribution system

Proposed Question: RO Question # 50

The plant is at rated conditions with all equipment operable when the following alarms are received on panel C3:

- "B" 125V DC UNDERVOLTAGE (C3RC-B7)
- "B" 125V DC GROUND (C3RC-D7)
- EDG "B" "GENERATOR BKR TRIP/INOP" (C3RC-B3)
- AUX/SU XFMR A-6 SUPPLY TRIP (C3LC-E2)

The reactor is about to manually scrammed.

Assuming no other operator action is taken, the status of the 4KV Emergency Buses one minute after the scram will be:

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

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: A-6 is de-energized. Plausible if the candidate does not recognize the symptoms of a D-17 loss. Alternatively the candidate may not realize the impact on the Startup Transformer supply breaker following the loss of D-17 (loss of control power).

- B. Incorrect: A-6 is de-energized. Plausible if the candidate does not recognize that the Shutdown Transformer supply is also impacted by the DC loss.
- C. Correct: The alarms are symptomatic of a loss of 125 VDC bus D-17. This will result in a complete loss of control power to bus A6. When the reactor is scrammed and the turbine trips bus A6 will de-energize. Bus A5 which still has control power, will fast transfer to the Startup Transformer.
- D. Incorrect: Bus A5 will fast transfer to the Startup Transformer. Plausible if the candidate believes that the Startup Transformer transfer logic is disabled by the loss of D-17.

Technical Reference(s): PNPS 5.3.12, Attachment 1, (Attach if not previously provided)  
 Caution 3.

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-01-02 EO 8 (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  
 Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7  
 55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295038	EA2.04
	Importance Rating	4.1	

Ability to determine and/or interpret the following as they apply to HIGH OFF-SITE RELEASE RATE : Source of off-site release

Proposed Question: RO Question # 51

The plant is operating at full power with the AOG system aligned for normal full power operation. The following indications are observed:

- Pre Treat Rad Monitors are steady
- Post Treat Rad Monitors are increasing
- Main Stack Rad Monitors are increasing
- Normal Recombiner temperatures for full power operation
- Alarm ADSORBR VESSEL TEMP HI, CP600R-B2, alarms

Which one of the following could result in these indications?

- A. Low steam dilution flow
- B. High steam dilution flow
- C. Wetting of the Recombiner
- D. Wetting of the Charcoal Adsorbers

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: Low steam dilution flow would result in higher than normal recombiner temperatures.
- B. Incorrect: High dilution flow would result in abnormal recombiner temperature.
- C. Incorrect: This would result in lower recombiner temperatures.
- D. Correct: IAW PNPS 2.2.106, Attachment 9, item H, charcoal performance will deteriorate when the charcoal gets wet. Holdup times for krypton and xenon will

decrease and plant emissions will increase. The Post Treat Monitors and the Mani Stack Monitors are both downstream of the Adsorbers.

Technical Reference(s): PNPS 2.2.106, Attachment 9, item (Attach if not previously provided)  
H  
PNPS 2.4.136, Section 5.0,  
discussion items [1] and [2]  
PNPS 2.4.141, Section 5.0,  
discussion item [1]

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-04-11, EO 3k. 9d (As available)

Question Source:	Bank #	Made editorial and format changes.
	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 #	1	
	Group #	1	
	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 is at power with the following initial Main Generator Parameters:

- MWe loading: 680
- MVAR loading: 100 lagging
- Hydrogen Gas pressure: 45 psig
- Main Generator Voltage Regulation: Auto

Then, a disturbance on the grid causes grid voltage to slowly lower over time.

Given the expected response of the Main generator to the above, when will the FIRST Main Generator Limit be exceeded? Note: The Generator Capability Curve is provided for your use.

When MVARs

- A. rise to 170 MVARs lagging
- B. rise to 360 MVARs lagging
- C. lower to 100 MVARs leading
- D. lower to 180 MVARs leading

Proposed Answer: A

Explanation (Optional):

- A. 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. This will cause the MVAR loading to increase in the lagging direction. The 45 psig hydrogen pressure curve will be exceeded when MVAR loading exceeds 170 MVARs.

- B. Incorrect: A limit was first exceeded when MVAR loading exceeds 160 MVARs.
- C. Incorrect: MVARs will increase in the lagging direction as the generator picks up reactive loading as grid voltage lowers. This is plausible if the candidate believes the opposite and applies the 100 MVAR leading limit of PNPS 2.2.2 (100 MVAR leading).
- D. Incorrect: MVARs will increase in the lagging direction This is plausible if the candidate believes the opposite and using the 45 psig hydrogen pressure limit. If so the limit will be exceeded when MVARs lower to 160, leading.

Technical Reference(s): 2.4.144, Attachment 4 (Attach if not previously provided)  
 2.2.2 Precautions 4 and 5  
 MOTORS AND GENERATORS  
 LP

Proposed References to be provided to applicants during examination: Attachment 4 of PNPS 2.4.144.

Learning Objective: O-RO-02-01-08, EO17f (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 5  
 55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030	EA2.01
	Importance Rating	4.1	

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Suppression pool level  
Proposed Question: RO Question # 53

Which ONE of the following corresponds to the maximum torus water level at which the HPCI System must be secured and injection prevented irrespective of adequate core cooling?

- A. 95 inches
- B. 90 inches
- C. 85 inches
- D. 50 inches

Proposed Answer: A

Explanation (Optional):

- A. Correct: IAW EOP-3, HPCI must be secured at 95” because the turbine exhaust will be above the water level in the torus and this can lead to over pressurizing the primary containment.
- B. Incorrect: IAW – EOP-3, this torus level uncovers the Downcomers and require E-Depress.
- C. Incorrect: This value is too low and will result in HPCI pressurizing the primary containment.
- D. Incorrect: This is the value associated with uncovering the SRV tail pipes.

Technical Reference(s): EOP-3, Steps TL-12 & 13 (Attach if not previously provided)



Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTSI # 1459  
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 9  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295018	2.4.47
	Importance Rating	4.2	

Emergency Procedures / Plan: Ability to diagnose and recognize trends in an accurate and timely manner utilizing the appropriate control room reference material. (Partial or Total Loss of CCW)

Proposed Question: RO Question # 54

The plant is at full power with a normal configuration on the Salt Service Water System.

The control room crew notes the following changes in SSW and RBCCW indications:

	Before	After
SSW loop "A" pressure as read by PI-3828:	30 psig	31 psig
SSW loop "B" pressure as read by PI-3829:	30 psig	31 psig
SSW loop "A" flow as read by FI-6240:	500 gpm	530 gpm
SSW loop "B" flow as read by FI-6241:	1750 gpm	1100 gpm

Using the partial SSW P&ID provided, which one of the following is consistent with the above changes in SSW parameters?

- A. "B" Loop SSW pump impeller failure
- B. Fouling of the "B" RBCCW heat exchanger
- C. Fouling of the "B" TBCCW heat exchanger
- D. Leak on the outlet of the "A" RBCCW heat exchanger

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The discharge of all SSW pumps is tied together. An impeller failure would reduce the pressure and flow in both loops.
- B. Correct: Fouling of the "B" RBCCW HX would reduce the flow in the "B" loop. This would cause the flow in the "A" loop to go up. Since the loops are tied together the pressure would be the same in both loops.

- C. Incorrect: Fouling of the "B" TBCCW HX would cause the flow to go up in both loops because the flow indicators are on the inlets of the two RBCCW HXs (see drawing).
- D. Incorrect: A leak on the outlet of the "A" RBCCW HX would cause flow in the "A" loop to go up as shown because the flow indicator is on the inlet to the HX. This would also cause "B" loop flow to go down. However pressure in the system would lower not rise.

Technical Reference(s): P&ID M212

(Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Partial SSW P&ID, showing just HX's and CR indications. White-out anything doing with automatic actions such as Load shed on SSW valves.

Learning Objective: O-RO-02-02-02, Terminal Objective, (As available)  
Respond to a loss of one SSW loop.

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 4  
55.43

Secondary coolant and auxiliary systems that affect the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005	2.4.21
	Importance Rating	4.0	

Emergency Procedures / Plan: Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc. (Main Turbine Generator Trip)

Proposed Question: RO Question # 55

Following a Main Turbine Trip from rated conditions EPIC displays the following indication:

- CALLRODS indicates “YES”

The CALLRODS indication means that:

- A. All control rods are at least at position 02.
- B. The program has activated but has not yet determined rod status.
- C. No more than one rod is fully withdrawn and all other control rods are at least at position 04.
- D. No more than one control rod is fully withdrawn and all other control rods are fully inserted.

Proposed Answer: A

Explanation (Optional):

- A. Correct: Upon receipt of a Scram signal, the CALLRODS program is triggered by backup Scram valve relay 5A-K21B. After a 25-second time delay to allow the control rods to settle, the program evaluates control rod position by evaluating whether the 00 or 02 position switch is currently made up. Any control rods which do not satisfy this criteria are stored for further evaluation in remaining phases. If all 145 control rods satisfy the criteria, the CALLRODS point changes status to NML YES. Although there are other phases to the program all that can be said following a “YES” indication is that all rods are at least at position 02.
- B. Incorrect: The YES indication means that the program has completed its run and determines that all rods are to or beyond position 02. Plausible if the candidate is not

familiar with the program nomenclature but understands that the program may take up to 3 minutes to determine rod status.

- C. Incorrect: This condition would result in a "NO" indication. Plausible in that one rod being fully withdrawn would satisfy TS shutdown margin and position 04 is the Maximum Subcritical Banked Withdrawal Position.
- D. Incorrect: This condition would satisfy shutdown margin.

Technical Reference(s): PNPS 2.2.89, Discussion item [6] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-04, EO-4.a, 4.b (As available)

Question Source: Bank # PNPS 3133  
Modified Bank #  
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 #	1	
	Group #	1	
	K/A #	295006	2.2.42
	Importance Rating	3.9	

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications (SCRAM).

Proposed Question: RO Question # 56

Which one of the following instrument failures would require an Active LCO as specified in Tech Specs and PNPS 1.3.34.2, Limiting Conditions for Operation Log?

Assume all other instruments are operable.

- A. IRMs "A" and "C" are inoperable with reactor power at 100%.
- B. APRM "A" is inoperable with the reactor critical at 200 cps in the source range.
- C. One reactor high pressure scram switch is inoperable with the Reactor Mode Switch in Startup.
- D. One SDIV high level scram switch is inoperable with the Reactor Mode Switch in Shutdown.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: IRM's are bypassed and not required when APRM's are on scale and the reactor mode switch is in the Run position. Plausible in that 3 of the 4 IRMs per trip system are required when the IRMs are required to be operable. This would warrant a Tracking LCO per PNPS 1.3.34.2.
- B. Incorrect: APRMs are required to be operable with the Mode Switch in Startup. With the reactor critical the Mode Switch is in Startup. However only 2 of the 3 available APRMs per trip system are required. This would warrant a Tracking LCO per PNPS 1.3.34.2.
- C. Correct: An active LCO is required when the inoperable component or system is required by the Technical Specifications for the current plant condition. Per TS Table 3.1.1, all reactor high pressure scram switches are required to be operable with the Mode Switch in Startup.

D. Incorrect: No scram functions are required when the Mode Switch is in Shutdown. Plausible in that this function is required in all mode switch positions other than Shutdown with some caveats while in Refuel. This would warrant a Tracking LCO per PNPS 1.3.34.2.

Technical Reference(s): PNPS 1.3.34.2 section 3.0, items [1] and [12] (Attach if not previously provided)  
TS Table 3.1.1 and associated notes

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-07, EO 23 (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 10  
55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031	EA2.03
	Importance Rating	4.2	

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER

LEVEL : Reactor pressure

Proposed Question: RO Question # 57

Following a trip of "A" Recirc Pump, the plant is operating with the "B" Recirc Pump in service.

Then a complete loss of feed occurs. Plant conditions are as follows:

- Current Drywell Pressure: 1.4 psig, slowly lowering
- Current RPV level: -30 inches, slowly rising
- HPCI has just been placed in pressure control
- Current RPV Pressure: 945 psig and lowering slowly
- Lowest RPV level during transient: - 80 inches
- Lowest RPV Pressure during transient: 920 psig

The following RHR indications are also observed:

- MO-1001-28A, LPCI INJ THROTTLE VLV #1: Open
- MO-1001-29A, LPCI INJ VLV #2: Closed
- MO-1001-28B, LPCI INJ THROTTLE VLV #1: Open
- MO-1001-29B, LPCI INJ VLV #2: Closed

Which one of the following statements is correct regarding LPCI Loop Select? Assume no operator actions have been taken relative to RHR/LPCI.

LPCI Loop select ...

- A. should have closed MO-1001-28A
- B. should have closed MO-1001-28B
- C. will close MO-1001-28A during the subsequent plant cooldown
- D. will close MO-1001-28B during the subsequent plant cooldown

Proposed Answer: C



Explanation (Optional):

- A. Incorrect – LPCI Loop select activated when water level lowered below -46 inches. Normally it will select a loop shortly after activating by examining the Dp across the jet pump risers. If it cannot discern a sufficient difference it will default to loop “B”. In single loop however (or no pumps running), it will wait till RPV pressure lowers below 900 psig to select a loop. Since pressure is 945 psig this has not yet occurred. The closure signal will be sent to 28A.
- B. Incorrect – LPCI Loop select activated when water level lowered below -46 inches. Normally it will select a loop shortly after activating by examining the Dp across the jet pump risers. If it cannot discern a sufficient difference it will default to loop “B”. In single loop however (or no pumps running), it will wait till RPV pressure lowers below 900 psig to select a loop. Since pressure is 945 psig this has not yet occurred.
- C. Correct – LPCI Loop select activated when water level lowered below -46 inches. Normally it will select a loop shortly after activating by examining the Dp across the jet pump risers. If it cannot discern a sufficient difference it will default to loop “B”. In single loop however (or no pumps running), it will wait till RPV pressure lowers below 900 psig to select a loop. Since pressure is 945 psig this has not yet occurred. When pressure lowers to 900 psig it will select a loop. Because the transient was initiated by a loss of feed the riser dps will be the same and the “B’ loop will be selected. This will send a closure signal to MO-1001-28A.
- D. Incorrect – Because the transient was initiated by a loss of feed the riser dp’s will be the same and the “B’ loop will be selected. This will send a closure signal to MO-1001-28A.

Technical Reference(s): 2.2.19, Attachment 9

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective:

(As available)

Question Source: Bank # LOR Bank #17

Modified Bank #

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 #	1	
	Group #	1	
	K/A #	295001	2.2.25
	Importance Rating	3.2	

Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (Partial or Complete Loss of Forced Core Flow Circulation)

Proposed Question: RO Question # 58

The plant is at rated conditions.

Then a malfunctioning "B" Recirc Flow controller results in a lowering "B" Recirc pump speed. Additional information is as follows:

- The reactor operator inserts a scoop tube lock on the "B" Recirc MG set and the speed reduction is terminated.
- Reactor power stabilizes at 85%.
- "B" Recirc pump speed stabilizes at 68%
- "A" Recirc pump speed is 80%

(1) Has the Tech Spec Recirc Speed Mismatch LCO been exceeded and

(2) What are the bases for this TS limitation?

- A. (1) Yes  
(2) Minimize Jet Pump Vibration
- B. (1) No  
(2) Minimize Jet Pump Vibration
- C. (1) Yes  
(2) Ensure proper functioning of the LPCI Loop Selection logic
- D. (1) No  
(2) Ensure proper functioning of the LPCI Loop Selection logic

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The limit is based on ensuring the proper functioning of the LPCI Loop

Selection logic.

- B. Incorrect: The limit is a 10% speed mismatch when power is above 80%. Therefore the limit was exceeded when pump speed lowered below 70%. Plausible if the candidate applies the limit of 15% which is only applicable below 80% power. Additionally, the limit is based on ensuring the proper functioning of the LPCI Loop Selection logic. There is a concern with excessive vibration but this is addressed by the minimum pump speed limits which protect against cavitation (see PNPS 2.2.84, Precaution 4).
- C. Correct: The limit is a 10% speed mismatch when power is above 80%. Therefore the limit was exceeded when pump speed lowered below 70%. As discussed in the TS bases for TS 3.6.F.1 the licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. At or below 80% power the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise.
- D. Incorrect: The limit is a 10% speed mismatch when power is above 80%. Therefore the limit was exceeded when pump speed lowered below 70%.

Technical Reference(s): TS 3.6.F.1 and associated bases. (Attach if not previously provided)  
PNPS 2.2.84, Precaution 4

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-02 EO 35f (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  
Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5  
55.43

Facility operating characteristics during steady state and transient conditions, including coolant chemistry, causes and effects of temperature, pressure and reactivity changes,

effects of load changes, and operating limitations and reasons for these operating characteristics.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295014	AK1.01
	Importance Rating	3.7	

Knowledge of the operational implications of the following concepts as they apply to  
**INADVERTENT REACTIVITY ADDITION** : Prompt critical

Proposed Question: RO Question # 59

Control rods are being withdrawn for a reactor plant startup.

- The Rod Worth Minimizer is bypassed
- A second operator is verifying rod movement but several control rods are withdrawn out of sequence
- The reactor has just been taken critical
- Reactor power is 1000 CPS in the source range

Then, an uncoupled, fully inserted, center control rod blade falls to the full out position.

All other systems respond as designed.

Regarding the fuel bundles around the dropped rod, the transient will result in .....

- A. no cladding or fuel damage
- B. only minor cladding perforations
- C. instantaneous fragmentation and dispersal of the fuel rods
- D. complete melting of the fuel pellets but no significant cladding failure

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: Plausible if the operator believes that the combination of the control rod velocity limiter, fuel loading and the Reactor Protection System will prevent damage.
- B. Incorrect: Plausible in that per the Safety Analysis, if control rods are withdrawn in-sequence and a control rod drop accident should occur, the highest attainable heat generation is <220 cal./gm. Conservatively, this is estimated to result in perforating 330

fuel rods. However in this case multiple control rod errors have occurred.

- C. Correct: The worst case rod drop accident would be one which occurred in the startup region. Multiple operator withdraw and insert errors could establish a critical central core region with the center rod fully inserted. If this rod were then to suffer a rod drop accident, considerable fuel damage could result when the reactor goes prompt critical. Enthalpies greater than 425 cal/gram may result. This would produce instantaneous fragmentation and dispersal of the fuel rods. Pressure would increase explosively, and fission fragment release from the clad would be gross.
- D. Incorrect: Plausible in that 280 cal/gram is the design limit for the Rod drop accident. Although complete UO<sub>2</sub> melting does occur if the fuel enthalpy reaches 280 cal/gram, relatively few fission products are released from the clad and the resulting pressure transient is minimal. This is the bases for the Tech Spec limit.

Technical Reference(s): Rod Worth Minimizer reference (Attach if not previously provided)  
text, Attachment 1 (pages 29 & 30)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: NA

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

10 CFR Part 55 Content: 55.41 2  
55.43

General design features of the core, including core structure, fuel elements, control rods, core instrumentation, and coolant flow.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295033	EK2.02
	Importance Rating	3.8	

Knowledge of the operational implications of the following concepts as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Process radiation monitoring system

Proposed Question: RO Question # 60

Which one of the following alarms requires immediate entry into EOP-04?

- A. SBTG DISCH RAD HI, C904LC-F4
- B. REACTOR BLDG RAD HI, C904LC-A1
- C. REFUEL FLOOR RAD HI, C904LC-C7
- D. REACTOR BLDG VENT RAD HI C904LC-B5

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: This alarm in and of itself, does not require entry or re-entry into EOP-04. Plausible in that this alarm may occur if SBTG was in service while a primary system was discharging into the secondary containment.
- B. Incorrect: EOP-04 is entered on high area radiation levels as determined by local survey IAW PNPS 5.3.33, SECONDARY CONTAINMENT RADIATION SURVEYS FOR EOP-04. The entry condition does not utilize Hi Rad alarms associated with the Rx. Building.
- C. Incorrect: A Refuel Floor High radiation condition is not addressed by EOP-04.
- D. Correct: The EOP-04 entry is required if annunciator REACTOR BLDG VENT RAD HI C904LC-B5 alarms. This is because the alarm Indicates a release of radioactivity to the Reactor Building environment.

Technical Reference(s): EOP-04, Entry Conditions (Attach if not previously provided)



Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-06, EO 1 (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 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	
	K/A #	295029	EK3.01
	Importance Rating	3.5	

Knowledge of the reasons for the following responses as they apply to HIGH SUPPRESSION  
 POOL WATER LEVEL : Emergency depressurization  
 Proposed Question: RO Question # 61

EOP-03, Primary Containment Control requires that an Emergency Depressurization be performed when torus level cannot be restored and maintained below \_\_\_ (1) \_\_\_\_.

This is because \_\_\_ (2) \_\_\_\_.

- A. (1) 175 inches  
 (2) the torus to drywell vacuum breakers begin to cover at this value, increasing the likelihood of exceeding the negative design pressure of the drywell.
- B. (1) 175 inches  
 (2) torus level has reached the SRV Tail Pipe Level Limit (STPLL)
- C. (1) 300 inches  
 (2) the torus to drywell vacuum breakers begin to cover at this value, increasing the likelihood of exceeding the negative design pressure of the drywell.
- D. (1) 300 inches  
 (2) the torus level has reached the SRV Tail Pipe Level Limit (STPLL)

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The reason is because the STPLL is being exceeded. See below. Plausible in that there is an action in EOP-03 that addresses covering up the torus to drywell vacuum breakers at 180 inches.
- B. Correct: Step TL-7, directs that when torus level cannot be restored and maintained below 175 inches then an Emergency Depress is to be performed. This value corresponds to the SRV Tail Pipe Level Limit (STPLL). The STPLL is the lower of either:

(1) The Maximum Pressure Suppression Primary Containment Water Level

- (MPSPCWL, 175 in.) OR
- (2) The highest torus water level at which opening an SRV will not result in exceeding the allowable stresses in the SRV tailpipes, tailpipe supports, quenchers, or quencher supports.

The MPSPCWL (175 in.) is more limiting and defines the PNPS STPLL.

The STPLL (MPSPCWL) is therefore the water level corresponding to the bottom of the ring header in the torus. It is the upper torus water level bound that ensures pressure suppression capability of the primary containment during emergency depressurization through SRVs.

- C. Incorrect: The limit for Emergency Depress is 175 inches. Plausible in that there is an action in EOP-03 that addresses covering up the torus to drywell vacuum breakers at 180 inches. If a candidate thinks that the vacuum breakers tap off the top of the torus, this would lead him/her to select this answer.
- D. Incorrect: The limit for Emergency Depress is 175 inches.

Technical Reference(s): EOP-03 lesson plan, page 33 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-02, EO 27 (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 X  
Comprehension or Analysis

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 #	2	
	K/A #	295012	AA1.01
	Importance Rating	3.5	

Ability to operate and/or monitor the following as they apply to HIGH DRYWELL

TEMPERATURE : Drywell ventilation system

Proposed Question: RO Question # 62

The plant is at rated conditions with the drywell cooling fans aligned as follows:

- Four fans are in Standby
- All other fans are in Run

Then a small steam leak results in rising drywell temperature and pressure and the reactor scrams. Five minutes later, plant conditions are as follows:

- Drywell Temperature: 220 degrees and rising
- Drywell Pressure: 4.5 psig and rising
- Startup Transformer: Locked Out
- Bus A5: Powered by "A" EDG
- Bus A6: Powered by "B" EDG
- No operator action has been taken regarding the drywell cooling system

The drywell cooling fans that were originally in Run will be \_\_\_ (1) \_\_\_.

The drywell cooling fans that were originally in Standby will be \_\_\_ (2) \_\_\_.

- A. (1) Running  
(2) Off
- B. (1) Off  
(2) Running
- C. (1) Running  
(2) Running
- D. (1) Off  
(2) Off

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: None of the fans will be running. Plausible if the candidate believes that the fans originally in Run will restart when the EDGs re-energize the bus and since they restarted there is no need for the standby fans to start on low flow.
- B. Incorrect: None of the fans will be running. Plausible if the candidate believes that the standby fans will auto start on low flow after the EDGs re-energize their busses after the running fans tripped on loss of power.
- C. Incorrect: None of the fans will be running. Plausible if the candidate recalls that the standby fans auto start when the Unit Aux Feeder breakers trip open following a scram after a 45 second time delay.
- D. Correct: The EDGs supplying the bus during a high drywell pressure condition will generate load shed signals and trip all the drywell cooling fans supplied by the EDGs and prevent any auto starts.

Technical Reference(s): PNPS 2.2.49, 9 (Attach if not previously provided)  
PNPS 2.4.16, Attachment 8, sheet  
2

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-04, EO 8c (As available)

Question Source: Bank #  
Modified Bank # (Note changes or attach parent)  
New 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 #	1	
	Group #	2	
	K/A #	295034	EA2.02
	Importance Rating	3.7	

EA2.02 - Ability to determine and/or interpret the following as they apply to SECONDARY CONTAINMENT VENTILATION HIGH RADIATION : Cause of high radiation levels  
Proposed Question: RO Question # 63

The plant is at 2% power and starting up with Primary Containment inerting in progress IAW PNPS 2.2.70, Primary Containment Atmosphere Control System.

Annunciator C904LC, REACTOR BLDG VENT RAD HI now alarms.

Which one of the following potential sources could result in this alarm?

- A. Recirc Pump seal failure
- B. Reactor Feed pump seal failure
- C. Steam leak in the condenser compartment
- D. High activity on the discharge of the Mechanical Vacuum Pump

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: Although the drywell is being vented this would not result in an increase in REACTOR BLDG VENT rad levels because SBGT is used while inerting the containment. SBGT discharges to the main stack. Plausible if the candidate believes that inerting is done using the reactor building ventilation exhaust fans which is physically possible.
- B. Incorrect: The RFP ventilation would exhaust the contaminants out the roof exhausters.
- C. Correct: The Turbine Building Ventilation system supplies the condenser compartment. This system discharges to the Reactor Building Vent via the Turbine Building Exhaust Fans. This would then be seen on the Rx Building Vent PRMs.
- D. Incorrect the Mechanical Vacuum pump discharges to the Main Stack

Technical Reference(s): Figure 4, Turbine Building Ventilation Reference Text (Attach if not previously provided)  
Figure 1, Reactor Building Ventilation Reference Text

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

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

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 11  
55.43

Purpose and operation of radiation monitoring systems, including alarms and survey equipment.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295036	2.2.14
	Importance Rating	3.9	

Equipment Control: Knowledge of the process for controlling equipment configuration or status.  
 Secondary Containment High Sump/Area Water Level  
 Proposed Question: RO Question # 64

A fuel damaging event has resulted in EOP entry. Plant conditions are as follows:

- ATWS conditions exist
- SBLC is not available
- Many control rods have not inserted and control rods are being inserted using the RMCS
- A leak has been reported on the discharge of the only available CRD pump
- The CRD Quad is being pumped down and water level in the quad is currently stable at 4 inches
- The event has just been upgraded to a General Emergency based on the extent of fuel damage

Which of the following is correct regarding EOP-04, Secondary Containment Control actions?

- A. Leave the CRD pump in service.  
No other action is required.
- B. Secure and isolate the leaking CRD pump.  
No other action is required.
- C. Leave the CRD pump in service.  
Open the Rx Building Floor and Equipment Sump pump breakers.
- D. Secure and isolate the leaking CRD pump.  
Open the Rx Building Floor and Equipment Sump pump breakers.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The Rx Building Floor and Equipment Sump pump breakers should be opened.



- B. Incorrect: Per step SC-12, systems are not isolated if they are required to be operated by EOPS. Without SBLC, CRD is required to achieve shutdown status. Plausible in that otherwise, it would be isolated since the level in the quad is above the Max Normal Operating Value.
- C. Correct: IAW the 4th override in EOP-04, if plant conditions indicate that substantial core damage is occurring, the operator is directed to isolate the Reactor Building Sump pumps. If the Rx building sumps could be contaminated, opening the breakers for the RX building sump pumps will preclude pumping highly contaminated water outside the secondary containment. Per step SC-12, systems are not isolated if they are required to be operated by EOPS. Without SBLC, CRD is required to achieve shutdown status.
- D. Incorrect: Per step SC-12, systems are not isolated if they are required to be operated by EOPS. Without SBLC, CRD is required to achieve shutdown status. Plausible in that otherwise, it would be isolated since the level in the quad is above the Max Normal Operating Value.

Technical Reference(s): EOP-04, 4<sup>th</sup> override and step SC- (Attach if not previously provided)  
 12  
 EOP -04 LP pages 13, 17, 18

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-06 EO 3 and TO #7 (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 10  
 55.43

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295015	AK1.03
	Importance Rating	3.8	

Knowledge of the operational implications of the following concepts as they apply to INCOMPLETE SCRAM : Reactivity effects  
Proposed Question: RO Question # 65

Following a reactor scram, the following conditions are present:

- Four (4) rods are at position 04
- One (1) control rod is at position 20
- All other control rods are fully inserted
- Reactor Power is on IRM range 1 and lowering
- No Boron has been injected

Which one of the following is correct regarding reactor shutdown status and whether a plant cooldown is permitted?

- A. The reactor is not currently shutdown. A plant cooldown is not permitted.
- B. Current reactor shutdown status cannot be determined without reactor engineering support. A plant cool down is not permitted.
- C. The reactor is currently shutdown and will remain shutdown under all conditions if a reactor plant cool down is conducted. A plant cooldown is permitted.
- D. The reactor is currently shutdown but may restart if a reactor plant cool down is commenced. A plant cooldown is permitted, while monitoring for a reactor restart.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: As defined in PNPS 5.3.35 definition [40] the reactor is shutdown when subcritical with Reactor power below the heating range (on scale on IRM range 7 or below). Plausible if the candidate confuses the definition of shutdown and shutdown under all conditions. The reactor is not shutdown under all conditions because not all rods are inserted to position 04 or beyond.

- B. Incorrect: Definitions for shutdown and shutdown under all conditions are provided in PNPS 5.3.35 which allow an operator to make this determination without RE support. Plausible in that 5.3.35 does recognize that an RE could also make these determinations for conditions not specified in 5.3.35.
- C. Incorrect: The reactor is not shutdown under all conditions because not all rods are inserted to position 04 or beyond. Although a cooldown can be commenced the operator must monitor for a restart due to the reactivity effects of the cooldown.
- D. Correct: As defined in PNPS 5.3.35 definition [40] the reactor is shutdown when subcritical with Reactor power below the heating range (on scale on IRM range 7 or below). EOP-02 directs that if the reactor is shutdown and no boron has been injected then a reactor cooldown is to be conducted. If the reactor should subsequently go critical due to the effects of the cooldown, EOP-02 Override P-7 will terminate further cooldown.

Technical Reference(s): PNPS 5.3.35 (Attach if not previously provided)  
 EOP-02 steps P-6 thru P-8  
 EOP Lesson Plan, page 50

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-04, EO 16, (As available)

Question Source: Bank #  
 Modified Bank # LOR 310 Changed shutdown status which changed the answer  
 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 #		
	K/A #	G2.1.20	
	Importance Rating	4.6	

Ability to interpret and execute procedure steps.

Proposed Question: RO Question # 66

A system operating procedure is being performed. Under what conditions can procedure steps be performed out of sequence?

- A. When the steps are "bulleted".
- B. When the steps are "lettered".
- C. Only when supervisor concurrence is obtained.
- D. Only when the procedure authorizes specific steps to be performed out of order.

Proposed Answer: A

Explanation (Optional):

- A. Correct: IAW EN-HU-106, steps identified as bulleted steps can be performed in any sequence.
- B. Incorrect: Lettered steps must still be performed in sequence. Plausible in that the steps are not numbered.
- C. Incorrect: Supervisors cannot authorize procedure steps to be performed out of order. Plausible in that supervisors can provide concurrence that a step is not applicable.
- D. Incorrect: Bulleted steps can also be performed in any order regardless of whether there is explicit guidance in the procedure that certain steps can be performed out of order.

Technical Reference(s): EN-HU-106, section 5.2.2, item [3] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01, EO 129 (As available)

Question Source: Bank #  
Modified Bank #  
New X

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:

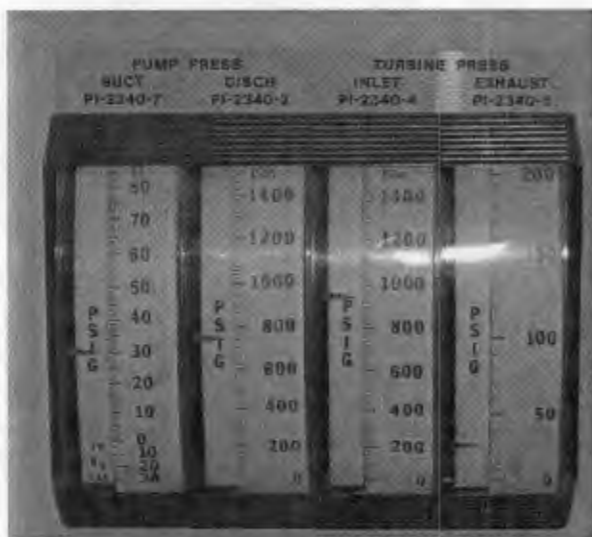
Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	_____	_____
	K/A #	G2.1.31	_____
	Importance Rating	4.6	_____

Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.

Proposed Question: RO Question # 67

During a loss of coolant accident in the drywell, HPCI automatically starts.

The operator notes that RPV level does not appear to be rising in response. The HPCI lineup on Panel C903 has been verified. Current HPCI indications are as shown below.



Based on the above indications which one of the following is correct

- A. The leak rate from the vessel must be approximately 4000 gpm.
- B. There is a leak on the HPCI discharge line.
- C. The HPCI pump impeller has failed.
- D. The HPCI controller has failed.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Pump discharge pressure is < than turbine inlet pressure so HPCI injection flow is not making it to the reactor. Plausible in that the flow on the HPCI controller is 4000 GPM.
- B. Correct: HPCI flow is 4000 GPM as called for by the controller. However Pump discharge pressure is < than turbine inlet pressure so HPCI injection flow is not making it to the reactor. Since the water is not going to the reactor it must be going somewhere – i.e., there must be a leak on the discharge line.
- C. Incorrect: If this were true the controller would not be showing 4000 GPM since the flow indicator is downstream of the pump.
- D. Incorrect: The controller is in Auto and set to control HPCI flow at 4000 GPM. The system is delivering 4000 GPM as indicated on the controller. Plausible if the candidate does not understand the indications provided by the controller.

Technical Reference(s): HPCI Reference Text Figure 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01 (As available)

Question Source: Bank # LOR Bank  
Modified Bank #  
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 #	3	
	Group #		
	K/A #	G2.2.40	
	Importance Rating	3.4	

Ability to apply technical specifications for a system.

Proposed Question: RO Question # 68

The plant is at rated conditions. Which of the following situations would render the "A" Salt Service Water System inoperable?

Situation 1: Average seawater temperature is 72 °F

Situation 2: 'A' RBCCW Heat Exchanger SSW Outlet Valve, MO-3800 jammed full open

Situation 3: 'A' TBCCW Heat Exchanger SSW Outlet Valve, MO-3801 jammed full open

- A. Situation 1 only
- B. Situation 3 only
- C. Situations 2 and 3 only
- D. Situations 1 and 2 only

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The Ultimate Heat sink (UHS) is operable up to a maximum seawater temperature of 75 °F
- B. Correct: Per Tech Spec bases, a SSW subsystem is considered OPERABLE when it has an OPERABLE UHS, two OPERABLE pumps with associated controls and instrumentation and the following valves on that subsystem operable:
  1. One TBCCW heat exchanger outlet valve unless the valve is throttled.
  2. One RBCCW heat exchanger outlet valve unless the valve is open.
 With 3801 jammed full open it cannot reposition to its accident position (10%) open.
- C. Incorrect: MO-3800 is full open and in its accident condition and therefore does not render the loop inoperable.



D. Incorrect: Neither the seawater temperature or MO-3800 being full open renders the loop inoperable (see above)

Technical Reference(s): Tech Spec Bases page B3/4.5, (Attach if not previously provided)  
pages 15 and 15a

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X  
Modified Bank #  
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 #	3	
	Group #		
	K/A #	G2.2.6	
	Importance Rating	3.0	

Knowledge of the process for making changes to procedures.

Proposed Question: RO Question # 69

While studying for an exam you discover a procedure error in the RHR system operating procedure.

In one section of the procedure, the RHR Loop "A" Torus Cooling/Spray Block Valve is identified as MO-1001-34B.

The procedure owner in Merlin is listed as "ODM" and approval authority has not been officially assigned to designees within the organization.

The procedure change generated to fix the error would be classified as a/an \_\_\_\_ (1) \_\_\_\_ (intent /non-intent).

The lowest level person in the Operations Department that can approve the change is \_\_\_\_ (2) \_\_\_\_.

- A. (1) intent  
(2) any SRO qualified shift manager
- B. (1) intent  
(2) the Operations Department Manager
- C. (1) non-intent  
(2) any SRO qualified shift manager
- D. (1) non-intent  
(2) the Operations Department Manager

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: IAW Attachment 6 of NOP 98A1, this change would be classified as a non-intent change. Plausible if the candidate believes non-intent changes are not allowed for safety related procedures. Plausible in that some types of changes for certain procedures are not allowed. For example EWN changes or Temporary Changes are not allowed for some 5.3 series procedures.

Additionally the change must be approved by the Operations Department Manager (ODM). Plausible in that a shift manager can approve an EWN change. However this change would not be associated with emergent work. If the candidate confuses the approval requirements, they would select this response.

- B. Incorrect: IAW Attachment 6 of NOP 98A1, this change would be classified as a non-intent change.
- C. Incorrect: The change must be approved by the Operations Department Manager (ODM).
- D. Correct: IAW Attachment 6 of NOP 98A1, this change would be classified as a non-intent change which states that one criteria for a non-intent change is "Changing equipment identifiers (for example: changing "B" pump to "A" pump when the Procedure section obviously is written for the "A" pump). All other criteria are also satisfied.

IAW section 6.1.3.1 of NOP 98A1, the procedure owner must approve all non-intent changes. In this case the ODM is the procedure owner.

Technical Reference(s): NOP98A1, Attachment 6 and sections 5.3 and 6.1.3.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01, EO 2b (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

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 #		
	K/A #	G2.3.4	
	Importance Rating	3.2	

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

Explanation (Optional):

- 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 References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # X  
Modified Bank #  
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 12  
55.43

Radiological safety principles and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.3.15	
	Importance Rating	2.9	

Knowledge of radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

K/A Justification: The C19 panels referred to in the stem contain fixed radiation monitors that will go into alarm on high particulate, iodine, gaseous activity or background radiation levels. Any monitor reaching an alarm setpoint will result in the Trouble Alarm mentioned in the stem.

Proposed Question: RO Question # 71

The plant is at rated conditions. SBGT is running and the torus is being vented via the 2 inch torus vents, AO-5041A and B, in order to adjust drywell to torus differential pressure.

While venting, annunciator C19 A/B Trouble, C904LC-B3 is received.

Which one of the following is correct?

- A. Direct Radwaste to pump the drywell sumps. Venting may continue while drywell leakage rates are being verified to be within limits.
- B. Direct Chemistry to verify the alarm locally. Venting may continue while the alarm is being verified.
- C. Venting may continue provided the SBGT Disch Rad Hi alarm is not received.
- D. Venting must be immediately secured.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: Venting must be immediately secured. Plausible in that venting may recommence if containment leakage rates are within Tech Spec limits.
- B. Incorrect: Venting must be immediately secured. Plausible in that there is no direct readout of the C-19 Rad monitors in the control room.

- C. Incorrect: Venting must be immediately secured. Plausible in that venting is being performed using SBTG.
- D. Correct: IAW PNPS Attachment 3, venting must be immediately secured.

Technical Reference(s): PNPS 2.2.70, Attachment 13, (Attach if not previously provided)  
 pages 4 and 5  
 ARP C904LC-B3

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-02, TO VENT THE (As available)  
 PRIMARY CONTAINMENT DURING  
 STATION OPS TO MAINTAIN  
 PRIMARY CONTAINMENT  
 PRESSURE WITHIN THE SPECIFIED  
 PRESSURE REGULATION BAND.

Question Source: Bank # PNPS 6280 Made several changes but the  
 question and answer remain the  
 same.  
 Modified Bank #  
 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 #		
	K/A #	G2.4.18	
	Importance Rating	3.3	

Knowledge of the specific bases for EOPs.

Proposed Question: RO Question # 72

Regarding EOP-03, Primary Containment Control:

Which one of the following is the bases for the requirement to initiate drywell sprays when torus bottom pressure exceeds 16 psig?

This ensures that...

- A. The differential pressure rating of the containment vents is not exceeded.
- B. Sufficient non condensable gases remain in the drywell to prevent chugging.
- C. The negative design pressure of the containment will not be exceeded when sprays are initiated.
- D. The reactor building to torus vacuum breakers do not lift resulting in oxygen being added to the primary containment.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: This is a function of the Primary Containment Pressure Limit
- B. Correct: Drywell sprays are initiated when torus bottom pressure exceeds 16 psig, the Suppression Chamber Spray Initiation Pressure (SCSIP), to preclude chugging - the cyclic condensation of steam at the downcomer openings.

If torus bottom pressure is greater than 16 psig, the possibility exists for "chugging" to occur. Chugging is the cyclic condensation of steam at the downcomer openings of the drywell vents. This cyclic condensation causes stresses at the Downcomers which can lead to fatigue failure of the Downcomers. This could cause any steam release in the drywell to bypass the torus and directly pressurize the torus airspace .

Chugging is a concern only if >99% of the non-condensibles (N<sub>2</sub>) in the drywell are in

the torus. The torus bottom pressure of 16 psig is based on the lowest pressure that the torus would experience if 95% of the drywell non-condensibles were in the torus. When drywell sprays are initiated, the resulting pressure reduction opens the torus-to-drywell vacuum breakers, drawing non-condensibles from the torus back into the drywell.

- C. Incorrect: This is a function of the Drywell Spray Initiation Limit.
- D. Incorrect: This is the bases for securing drywell sprays before containment pressure lowers to 0 psig.

Technical Reference(s): O-RO-03-04-05, Pages 40 and 41 (Attach if not previously provided)

Proposed References to be provided to applicants during examination:

Learning Objective: O-RO-03-04-02, EO-26 (As available)

Question Source: Bank # WTSI # 11647  
Modified Bank #  
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 9, 10  
55.43

Shielding, isolation, and containment design features, including access limitations.  
Administrative, normal, abnormal, and emergency operating procedures for the facility.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G2.4.29	
	Importance Rating	3.1	

Knowledge of the Emergency Plan

Proposed Question: RO Question # 73

You are a licensed RO on days off when the Emergency Plan is activated.

You have been contacted by the Emergency Plant Operations Supervisor (EPOS) and directed to support the Emergency Response Organization.

If the EPOS directs you to report and man the ENS, you will report to the \_\_\_ (1) \_\_\_.

If the EPOS directs you to report and man the PDP, you will report to the \_\_\_ (2) \_\_\_.

- A. (1) EOF  
(2) Control Room
- B. (1) EOF  
(2) TSC
- C. (1) Control Room  
(2) Control Room
- D. (1) Control Room  
(2) TSC

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The off-shift operator is directed to man the ENS in the control room. Plausible in that there is an ENS line in the EOF but it is manned by non-operations personnel.
- B. Incorrect: The off-shift operator is directed to man the ENS in the control room. Additionally the off-shift operator is directed to man the PDP in the control room. Plausible in that there is a PDP in the TSC but it is manned by non-operations personnel.

- C. Correct: IAW EPIP 210, the EPOS is required to call-out one operator to man both the ENS and the PDP in the control room.
- D. Incorrect: The off-shift operator is directed to man the PDP in the control room.

Technical Reference(s): EPIP-210, Control Room Augmentation, pages 4 and 5. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

Question History: Last NRC Exam: Not used

Question Cognitive Level: Memory or Fundamental Knowledge 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 #		
	K/A #	G 2.1.18	
	Importance Rating	3.6	

Ability to make accurate, clear and concise logs, records, status boards, and reports.

Proposed Question: RO Question # 74

A surveillance procedure is in progress that directs that an electrical lead be lifted during the performance of the surveillance.

Which of the requirements below must be met in order for the lifting of the lead NOT to be logged in the Lifted Lead and Jumper Log?

- Requirement 1: The procedure contains signoffs for the initial lifting and subsequent re-landing of the lead
- Requirement 2: The lifted lead is re-landed and returned to its normal configuration prior to the end of the Shift Manager's shift.
- Requirement 3: The surveillance results in an Active LCO and the LCO is tracked via the ESOMs software.

- A. Requirement 2 only
- B. Requirement 3 only
- C. Requirements 1 and 2
- D. Requirements 1 and 3

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The procedure must also contain signoffs for removal and re-landing of the lead.
- B. Incorrect: This is not a requirement. Plausible in that this is a configuration control process.
- C. Correct: An electrical lifted lead, jumper, or gagging device need not be entered in the LLJ Log if the activity is performed in accordance with an approved Station Procedure

provided that:

- The placing of the jumper, gagging device, lifting of the lead and the removal of the jumper, gagging device, landing of the lead are specific signoff steps in the approved governing Procedure.
- AND
- The jumper, gagging device, or lifted lead will be returned to its normal configuration prior to the end of the current SM working shift.

D. Incorrect: Requirement 3 is not required. Requirement 2 is.

Technical Reference(s): PNPS 1.5.9.1, page 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01, 18c (As available)

Question Source: Bank #  
Modified Bank #  
New X

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 #		
	K/A #	G2.2.15	
	Importance Rating	3.9	

Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.

Proposed Question: RO Question # 75

With the plant at 50% power, a closed motor operated valve located in the drywell must be tagged in the closed position.

Which one of the following is correct regarding the verification that the valve is in the closed position and tagged closed?

- A. A non-independent verification is performed by observing the indicated valve position in the control room by two operators. A non-independent verification that the correct breaker has been identified is performed before the breaker is opened.
- B. A non-independent verification is performed by observing the indicated valve position in the control room by two operators. The breaker for the valve is then opened and independently verified open.
- C. The valve is verified closed by observing the indicated valve position in the control room. The position indication is then independently verified. The breaker for the valve is then opened and independently verified open.
- D. The valve is verified closed by observing the indicated valve position in the control room. The position indication is then independently verified. A non-independent verification that the correct breaker has been identified is performed before the breaker is opened.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: An independent verification is required using the indirect method by observing the valve's position indication.
- B. Incorrect: An independent verification is required using the indirect method by observing the valve's position indication. Additionally, two operators must use concurrent verification that the correct breaker has been identified before opening the breaker.

- C. Incorrect: Two operators must use concurrent verification that the correct breaker has been identified before opening the breaker.
- D. Correct: Because the valve is in the drywell, the position of the valves is to be independently verified using the indirect method by observing the position indication in the control room. The opening of the correct breaker must be verified by two operators. PNPS 1.3.34 requires non-independent verification in circumstances where the action (opening a breaker) is irreversible.

Technical Reference(s): 1.3.34, Pages 42 & 43 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01 (As available)

Question Source: Bank # PNPS bank 3362 Reworded all distractors, replaced two  
 Modified Bank #  
 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 #		1
	Group #		1
	K/A #	295023	AA2.04
	Importance Rating		4.1

Ability to determine and/or interpret the following as they apply to REFUELING ACCIDENTS :  
Occurrence of fuel handling accident

Proposed Question: SRO Question # 76

The plant is refueling. The bridge is over the core with a fuel bundle loaded on the grapple. The grapple is in the full-up position.

Then, Refuel SRO reports a lowering pool level.

Additional indications are:

- Annunciator Spent Fuel Pool Level Lo (C903R-B2) has alarmed
- Annunciator Refuel Floor Rad Hi (C904LC-C7) has alarmed
  - ARM Spent Fuel Pool Area is determined to be the cause of the alarm
- No other Reactor Building ARMs are in alarm
- Standby Gas Treatment trains auto start

The SRO on the refuel floor is required to direct \_\_\_ (1) \_\_\_\_.

The SRO in the control room is required to direct \_\_\_ (2) \_\_\_\_.

- A. (1) the bundle be lowered into the nearest Reactor Vessel location with a fully inserted control rod  
(2) CRHEAFS be started
- B. (1) the bundle be lowered into the nearest Reactor Vessel location. The location need not contain a fully inserted control rod.  
(2) CRHEAFS be started
- C. (1) the bundle lowered into the nearest Reactor Vessel location with a fully inserted control rod  
(2) the reactor building be evacuated
- D. (1) the bundle be lowered into the nearest Reactor Vessel location. The location need not contain a fully inserted control rod.  
(2) the reactor building be evacuated

Proposed Answer: A

Explanation (Optional):

A. Correct: Both PNPS 5.4.3, Refuel Floor High Radiation and PNPS 2.4.31, Reactor Basin and/or Spent Fuel Pool Drain-Down are required to be entered. Both of these procedures require that IF a fuel bundle is grappled onto the Refuel Bridge mast, THEN IMMEDIATELY LOWER the fuel bundle into the nearest Reactor Vessel location with a fully inserted control rod OR the nearest Spent Fuel Pool location. Since the bridge is over the core the bundle should be lowered back into the vessel.

PNPS 5.4.3, subsequent actions, requires that the Control Room High Efficiency Air Filtration System (CRHEAFS) be started.

B. Incorrect: The vessel location must contain a control rod. Plausible if the candidate believes that there is always sufficient S/D margin available to tolerate one bundle being lowered into a cell without a control rod and that lowering the bundle takes precedence.

C. Incorrect: Both procedures require that the refuel floor be evacuated. Plausible in that PNPS 5.4.3, subsequent actions, does direct that the reactor building be evacuated under certain conditions. These are:

- Both SBTG trains failed
- Any Reactor Building ARM in alarm other than those on the refuel floor.

Neither of these conditions exists so the reactor building need not be evacuated. Since local actions may need to be taken to mitigate the lowering pool level, the reactor building should only be evacuated if necessary.

D. Incorrect: The fuel bundle is to be lowered into in the nearest Reactor Vessel location with a fully inserted control rod. Additionally, there is no requirement do evacuate the reactor building. Since local actions may need to be taken to mitigate the lowering pool level, the reactor building should only be evacuated if necessary.

Technical Reference(s): PNPS 5.4.3, sections 3.0 and 4.0 (Attach if not previously provided)  
PNPS 2.4.31, section 3.0

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-04-03, EOs 9a and 10a (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 7

Fuel handling facilities and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295030	EA2.04
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to LOW SUPPRESSION POOL WATER LEVEL : Drywell/ suppression chamber differential pressure: Mark-I&II  
Proposed Question: SRO Question # 77

The plant is at rated conditions with the following Primary Containment parameters.

- Torus water level is -6 inches
- Drywell to Torus differential pressure is 1.17 psid

If Torus water level CONTINUES to LOWER which one of the following is required by Tech Specs?

- Initiate an orderly shutdown and be in cold shutdown within 24 hours. No other Tech Spec action is required.
- Restore the drywell to torus differential pressure within the next 8 hours. No other Tech Spec action is required.
- Initiate an orderly power reduction to < 15% within the next 12 hours and be in cold shutdown within the next 24 hours.
- Initiate an orderly shutdown and be in cold shutdown within 24 hours AND restore the drywell to torus differential pressure within the next 8 hours.

Proposed Answer: D

Explanation (Optional):

- Incorrect: A differential pressure of 1.17 is also at the lower Tech Spec limit as specified by TS 3.7.A.8.a. If torus water level lowered further the drywell differential pressure would also lower. TS 3.7.A.8.c then requires that the differential pressure must be restored within the next 8 hours. If not then the reactor shall be less than 15% within the next 12 hours.
- Incorrect: A plant shutdown must also be initiated. This would be the correct answer if the candidate only applied the TS action for the out of spec differential pressure.

- C. Incorrect: This would be true if the candidate determined that the drywell to torus differential pressure could not be restored within 8 hours.
- D. Correct: IAW TS 3.7.A.1, the minimum Torus water level is 84,000 ft<sup>3</sup>. This corresponds to a torus level of -6 inches. This is also addressed in TS 3.7.A.1.m. If torus water level continued to lower both of these limits would be exceeded. TS 3.7.A.5 then requires that an orderly shutdown shall be initiated and the reactor shall be in a Cold Shutdown Condition within 24 hours.

A differential pressure of 1.17 is also at the lower Tech Spec limit as specified by TS 3.7.A.8.a. If torus water level lowered further the drywell differential pressure would also lower. TS 3.7.A.8.c then requires that the differential pressure must be restored within the next 8 hours. If not then the reactor shall be less than 15% within the next 12 hours.

Technical Reference(s): TS 3.7.A.1 (Attach if not previously provided)  
 3.7.A.8.a  
 TS Bases page B3/4.7-8

Proposed References to be provided to applicants during examination: TS 3.7.A.1 through A.8. Blackout sections 3.7.A.1.c through j, and 3.7.A.2.

Learning Objective: (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
 55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295031	EA2.02
	Importance Rating		4.2

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

Proposed Question: SRO Question # 78

A failure to scram has occurred and the following plant conditions exist:

- Reactor power is 20% and lowering
- RPV water level is +5 inches and lowering
- Both Recirc Pumps have been tripped
- RPV pressure is being controlled between 900 and 1050 psig with SRVs
- Drywell pressure is 3.2 psig
- Torus bulk water temperature is 112 °F and rising
- Injection has been terminated and RPV level is being intentionally lowered

IAW EOP-02 RPV Control Failure to Scram, which ONE of the following set of conditions would allow recommencing injection?

Power lowering to .....

- A. 4% with RPV level lowering to -15 inches
- B. 6% with RPV level lowering to -125 inches, actual
- C. 8% with RPV level lowering to -100 inches, actual
- D. 10% with RPV level lowering to -30 inches

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: Once lowering of level is commenced injection is not recommended until level is below -25 inches regardless of how much power has dropped. This is to prevent / mitigate power oscillations. Plausible in that power has dropped the BIIT curve (< 10%)

- B. Correct: Even though reactor power is still above 3%, injection is recommenced when RPV level drops reaches -125 inches in order to maintain adequate core cooling.
- C. Incorrect: Level must continue to be lowered because power is still above 3%. Plausible in that power has dropped below 10% which is the power level associated with exceeding the Boron Injection Initiation Temperature (BIIT) curve and exceeding the BIIT curve is a criterion for intentionally lowering level.
- D. Incorrect: Level must continue to be lowered because power is still above 3%. Plausible in that level is below -25 inches which is that RPV level associated with preventing or mitigating power oscillations. However the conditions presented require level to be lowered further to protect the containment.

Technical Reference(s): EOP-02, steps L-14 thru L-21 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EOP-02, "Q" leg only

Learning Objective: O-RO-03-04-04, 23g (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.

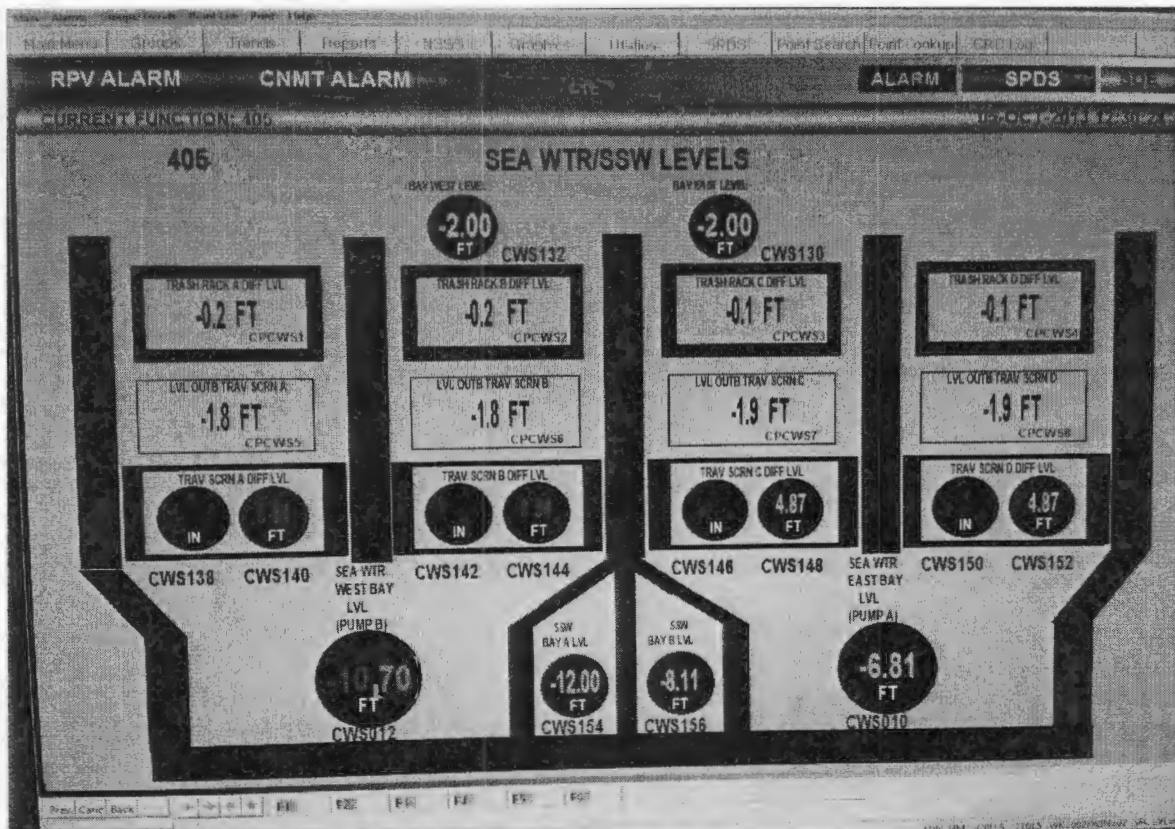
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295018	2.1.19
	Importance Rating		3.8

Conduct of Operations: Ability to use plant computers to evaluate system or component status.  
(Partial or Total Loss of CCW)

Proposed Question: SRO Question # 79

The plant is at rated power when a large mat of seaweed begins to clog the intake structure. Current conditions in the screenhouse are as shown below on EPIC screen 504.



Which one of the following is required?

- Insert a manual scram, enter PNPS 2.1.6, Reactor Scram and secure both seawater pumps.
- Enter PNPS 2.1.14, Station Power Changes, reduce power as required and then secure the "B" Seawater pump.



- C. Immediately secure the "B" seawater pump. Reduce power as required to maintain condenser vacuum IAW PNPS 2.4.36, Decreasing Condenser Vacuum. Securing SSW pumps is not required.
- D. Immediately secure the "B" seawater pump. Reduce power as required to maintain condenser vacuum IAW PNPS 2.4.36, Decreasing Condenser Vacuum and place the "A", "B" and "C" SSW pumps in Pull-to-Lock.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: A manual scram is not yet required as the "A" seawater pump has not yet reached a condition requiring that it be tripped or secured. Plausible in that it is approaching that level.
- B. Incorrect: The pump must be immediately tripped to preserve suction to the SSW pumps. Plausible in that there is a step to reduce power and then secure the pump. However level is too low to allow time to reduce power prior to tripping the pump.
- C. Correct: IAW PNPS 2.4.154, if any of the following Screenwell/Seawater Pump bay levels are observed, THEN IMMEDIATELY STOP the affected Seawater Pump in accordance with Step 4.0[4]: Screenwell/Seawater Pump bay level has dropped to below the -10 foot elevation. Tripping the SSW pumps is not required as they will retain suction down to -13 feet, 9 inches and level should quickly recover once the seawater pump is secured.
- D. Incorrect: Tripping the SSW pumps is not required as they will retain suction down to -13 feet, 9 inches and level should quickly recover once the seawater pump is secured.

Technical Reference(s): PNPS 2.4.154, section 4.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-03-23(07), TO RESPOND (As available)  
TO INTAKE STRUCTURE FOULING  
FROM THE CONTROL ROOM.

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

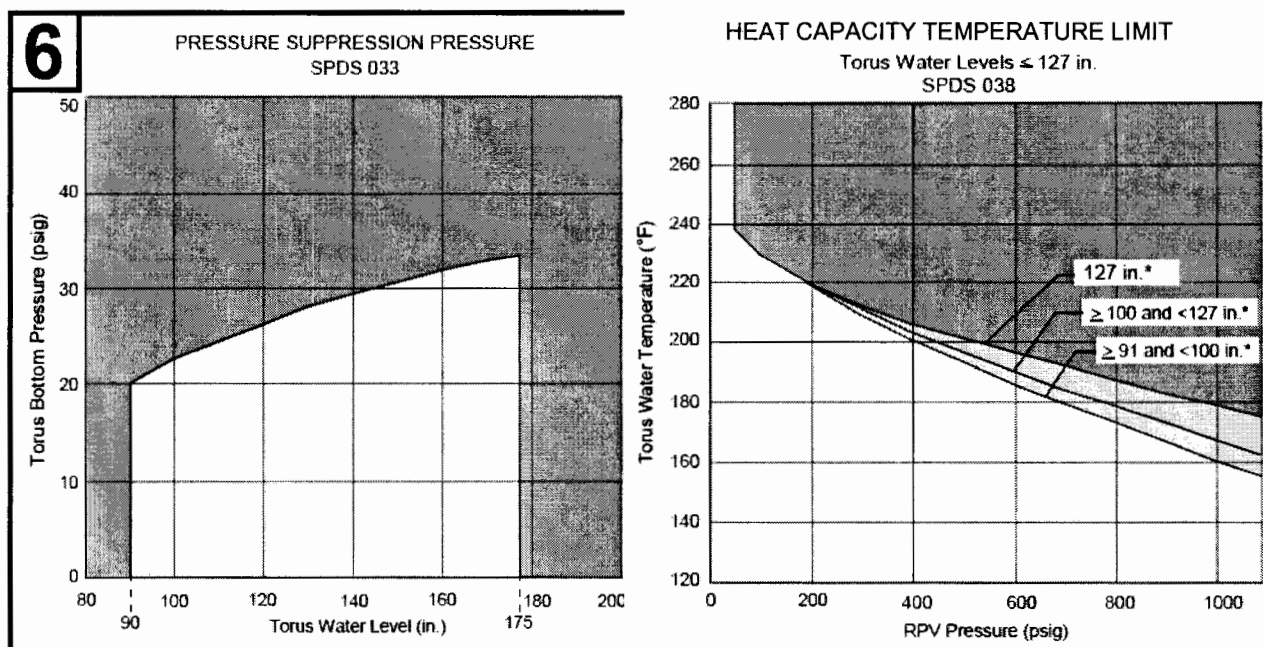
Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295025	2.2.44
	Importance Rating		4.4

Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives effect plant and system conditions. (High Reactor Pressure)

Proposed Question: SRO Question # 80

Which of the following sets of parameters would result in exceeding the Primary Containment Pressure Limit during an emergency depressurization?



- A. RPV pressure is 700 psig  
Torus water level is 89"
- B. Torus water level is 140"  
Torus bottom pressure is 31 psig
- C. Torus water temperature is 181°F  
RPV pressure is 700 psig  
Torus water level is 95"

- D. Torus water temperature is 160°F  
RPV pressure is 1050 psig  
Torus water level is 125"

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The PCPL will not be exceeded if an emergency depressurization (ED) is performed with these parameters. Plausible in that the Torus Downcomers are uncovered at this level and EOP-03 requires an ED at this value. However the SRV tail pipes are still covered at this value.
- B. Incorrect: The PCPL will not be exceeded if an emergency depressurization (ED) is performed with these parameters. Plausible in that the PSP is being exceeded at these values.
- C. Correct: The 3 parameters provided result in exceeding the HCTL.. The Heat Capacity Temperature Limit (HCTL) is the highest torus water temperature from which emergency RPV depressurization will not raise:

- Torus water temperature above the maximum temperature capability of the torus and equipment within the torus which may be required to operate when the RPV is pressurized, or
- Torus bottom pressure above the Primary Containment Pressure Limit (PCPL),

while the rate of energy transfer from the RPV to the containment is greater than the capacity of the containment vent.

For PNPS, the second criterion defines the HCTL.

- D. Incorrect: The HCTL is not being exceeded. Plausible if the candidate uses the lower limit for torus water  $\geq 91$  inches but  $< 100$  inches, in which case the candidate would think that the limit was being exceeded.

Technical Reference(s): O-RO-03-04-05, Page 29 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-05, EO9 (As available)

Question Source: Bank # WTSI Bank 3717  
Modified Bank #  
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 #		1
	Group #		1
	K/A #	295019	2.1.32
	Importance Rating		4.0

Conduct of Operations: Ability to explain and apply all system limits and precautions. (Partial or Total Loss of Inst. Air)

Proposed Question: SRO Question # 81

The reactor is at rated conditions, with the instrument air system aligned for normal full power operation.

Then, a loss of instrument air occurs. Procedure 5.3.8, Loss of Instrument Air is entered.

Which one of the following is correct regarding the 5.3.8 procedure limit below, the bases for that limit and the required action?

If Instrument Air Main Header pressure is less than 65 psig, then enter procedure ...

- A. 2.1.6 Reactor Scram and manually scram the reactor. The ADS accumulators will not have sufficient volume for the required number of actuations at this pressure.
- B. 2.1.6 Reactor Scram and manually scram the reactor. Control rods may begin drifting into the core at this pressure.
- C. 2.4.49, Feedwater Malfunctions and take manual control of the Feed Reg Valves in the condenser bay. The Feed Reg Valves will lock up at this pressure.
- D. 2.4.49, Feedwater Malfunctions and reduce Reactor Power to stabilize reactor water level IAW Section 7.11 of PNPS 2.1.14, "Station Power Changes". The Condensate Minimum Flow Valve will fail open at this pressure reducing available condensate flow.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: PNPS 5.3.8 requires a scram based on rods drifting in. Plausible in that the procedure addresses the ADS accumulators. However the pressure is where the ADS accumulators have insufficient volume is 91 psig. The required action is to close AO-4356, Essential Instrument Air Block Valve if instrument air is aligned to the drywell and pressure falls below this value. At rated conditions this would not be the case.

- B. Correct: IAW the Subsequent Action of procedure 5.3.8, the reactor is to be manually scrammed at this air pressure. This is because the scram air header pressure is lowering and the scram outlet valves may begin to open causing control rods to drift into the core.
- C. Incorrect: The procedure action at this pressure is to scram the reactor. Plausible in that the FRVs will lockup on lowering air pressure. However this will occur at 40 psig.
- D. Incorrect: The procedure action at this pressure is to scram the reactor. Plausible in that the Condensate Min Flow valve will be affected as pressure lowers. However the valve fails closed. If it were to fail open the impact would be as described.

Technical Reference(s): 5.3.8, subsequent actions (Attach if not previously provided)  
 ARP C905R-F1

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-02-28(01), Terminal (As available)  
 Objective, RESPOND TO A LOSS OF INSTRUMENT AIR.

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 X  
 Comprehension or Analysis

10 CFR Part 55 Content: 55.41  
 55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		1
	K/A #	295004	2.2.25
	Importance Rating		4.2

Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits (Partial or Total Loss of DC Pwr)

Proposed Question: SRO Question # 82

The plant is at rated conditions with 125 VDC Battery Backup Charger D14 inoperable. All other equipment is operable.

Then, the "B" 125 VDC Battery Charger fails. "B" Battery terminal voltage is later reported as having lowered to 110 VDC.

Assuming that conditions do not change the plant must be in Cold Shutdown in \_\_\_ (1) \_\_\_ hours.

This is because the battery is not capable supplying the required loads over the \_\_\_ (2) \_\_\_ hour service load period.

- A. (1) 24  
(2) 12
- B. (1) 72  
(2) 12
- C. (1) 24  
(2) 8
- D. (1) 72  
(2) 8

Proposed Answer: C

Explanation (Optional):

- A. Incorrect: The service load period is 8 hours.
- B. Incorrect: The plant must be cold shutdown within 24 hours. Plausible because Tech Spec 3.9.B.5 allows 3 days of continued operation if TS 3.5.F is met. However 3.5.F is not met.



Additionally, the service load period is 8 hours.

- C. Correct: A battery terminal voltage of equal to or greater than 119.0 volts for the 125 volt system under normal station load is capable of supporting bounding accident shutdown loads over the 8-hour service load period. The battery terminal voltage will decrease further with higher load levels but will still remain capable of performing its design function, based on the Pilgrim DC load flow studies results. Therefore the battery is inoperable.

IAW Tech Spec 3.9.B.5 from and after the date that one of the 125 or 250 volt battery systems is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding three days within electrical safety considerations, provided repair work is initiated in the most expeditious manner to return the failed component to an operable state, and Specification 3.5.F is satisfied. However TS 3.5.F.1 is not satisfied because the "B" EDG and the "B" side low pressure core and containment systems are inoperable. TS 3.5.F.1 requires that the plant be in cold shutdown in 24 hours if this LCO is not met.

- D. Incorrect: The plant must be cold shutdown within 24 hours. Plausible because Tech Spec 3.9.B.5 allows 3 days of continued operation if TS 3.5.F is met. However 3.5.F is not met.

Technical Reference(s): Tech Spec 3.9.B.5 and associated bases. (Attach if not previously provided)  
TS 3.5.F  
PNPS 5.3.12, Discussion item [1] and [2]

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-01-02, EO 15 and 16 (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 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295020	AA2.02
	Importance Rating		3.4

Ability to determine and/or interpret the following as they apply to INADVERTENT CONTAINMENT ISOLATION : Drywell/containment temperature  
Proposed Question: SRO Question # 83

The plant is at rated power when the following sequence of events occurs.

Time:

- T=0 RBCCW valve MO-4009A, RBCCW Loop B Nonessential Loop Inlet Valve is inadvertently closed
- T=+ 5min The reactor is manually scrammed when MO-4009A cannot be reopened
- T=+ 8 min Drywell pressure exceeds 2.2 psig
- T=+15 min Drywell temperature exceeds 250 °F
- T=+ 25 min Ability to reopen MO-4009A has been restored but the valve has not yet been reopened.

Given the above conditions which one of the following is the correct regarding:

(1) Whether RBCCW can or cannot be restored to the drywell

AND

(2) Any required actions?

- A. (1) Cannot  
(2) The only method to control drywell temperature is IAW EOP-03, Primary Containment Control.
- B. (1) Can  
(2) Restore flow by re-opening MO-4009A IAW Attachment 7, Opening RBCCW Valve MO-4009A of PNPS 2.4.42, Loss of RBCCW.  
No other action is required.
- C. (1) Can  
(2) Restore flow by performing Section 4.2 Recovery of RBCCW Loop "B" with an Elevated Drywell Temperature of PNPS 2.4.42, Loss of RBCCW.

- D. (1) Cannot
- (2) Defeat the Group 2 isolation signal and commence a venting and purging of the drywell with air IAW PNPS 2.4.44, Loss of Drywell Cooling.

Proposed Answer: C

Explanation (Optional):

- A. Incorrect. RBCCW flow can be restored. Plausible in that there are two parts to PNPS 2.4.42, Loss of RBCCW, Section 4.2 RECOVERY OF RBCCW LOOP "B" WITH AN ELEVATED DRYWELL TEMPERATURE. Both sections deal with drywell temperatures over 250 degrees. However the 1<sup>st</sup> section is utilized if the drywell temperature is also affiliated with a LOCA. In this case the RBCCW loop is recovered but flow is not restored to the drywell. If flow could not be restored the required actions would be as described in EOP-03.
- B. Incorrect: With RBCCW flow having been loss for > 6 minutes and drywell temperature being > 250 degrees, there is a potential for a water hammer event when flow is restored. The process of 2.4.42, Loss of RBCCW, Section 4.2 must also be followed.
- C. Correct: With RBCCW flow having been loss for > 6 minutes and drywell temperature being > 250 degrees, there is a potential for a water hammer event when flow is restored. The process of 2.4.42, Loss of RBCCW, Section 4.2 must be followed. MO-4009A must then be re-opened using Attachment 7 of this procedure.
- D. Incorrect: RBCCW flow can be restored if the process described in section 4.2 of 2.4.42 is followed. Also PNPS 2.4.44 does address venting the containment but there is no authorization to defeat the Group II isolation in that procedure to facilitate doing so.

Technical Reference(s): PNPS 2.4.42, section 4.2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-02-06, EO11 (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 #		2
	K/A #	500000	2.4.8
	Importance Rating		4.5

Emergency Procedures / Plan: Knowledge of how abnormal operating procedures are used in conjunction with EOP's (High CTMT Hydrogen Conc)

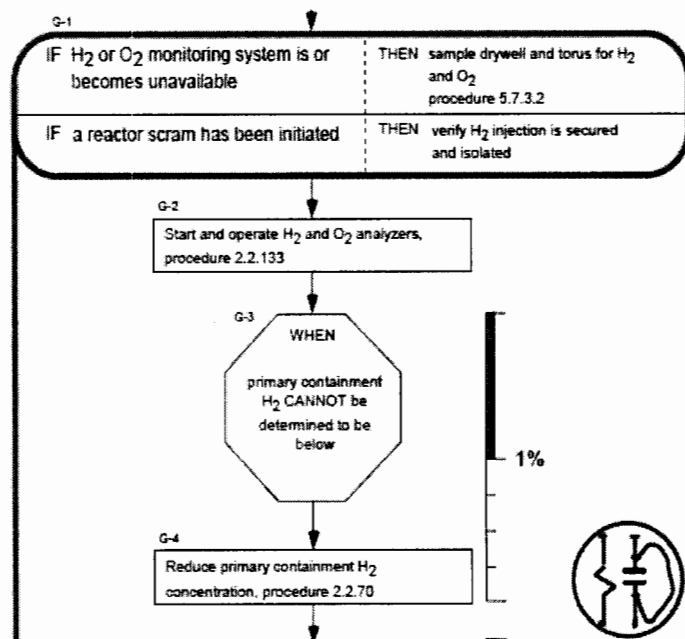
Proposed Question: SRO Question # 84

EOP-03, Primary Containment Control is being executed. Plant conditions are as follows:

- All control rods inserted.
- Reactor level: -70 inches, actual, and slowly rising via CRD injection.
- RPV Pressure: 375 psig lowering
- RPV level was below -150 inches, actual, for a period of time before beginning to recover
- H2 injection was isolated following the scram
- Neither H<sub>2</sub>O<sub>2</sub> analyzer can be placed in service

IAW PNPS 5.7.3.2, Drywell and Torus Atmospheric Sampling under Emergency Conditions and EOP-03, which one of the following is correct

- (1) regarding EOP-03 combustible gas control,  
AND  
(2) any required EOP-03 action?



- A. (1) Hydrogen concentration can be assumed to be < 1%  
(2) PASS sampling is not required.
- B. (1) Hydrogen concentration cannot be determined.  
(2) Direct a PASS sample be obtained before taking additional action.

- C. (1) It must be assumed that Hydrogen concentration is > 1%  
(2) Commence venting the containment IAW procedure 2.2.70, Primary Containment Control. Exceeding release limits is authorized.
- D. (1) It must be assumed that Hydrogen concentration is > 1%  
(2) Commence venting the containment IAW procedure 2.2.70, Primary Containment Control. Exceeding release limits is not authorized.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: Plausible in that RPV level is currently -70 inches and recovering. PNPS 5.7.3.2 also directs that if all the following conditions are met then PASS sampling is NOT required:
- The Reactor is shutdown and will remain shutdown under all conditions;
  - Initial oxygen concentration in the Drywell and Torus was 3% or less;
  - Only nitrogen is and was lined up to the Drywell pneumatic header;
  - The Torus-to-Reactor Building vacuum breakers have not opened since oxygen concentration was established at 3% or less;
- Since there is no evidence that any of these conditions were not met, PASS sampling is not required.
- B. Incorrect: It must be assumed that [H<sub>2</sub>] is > 1% since RPV level dropped below TAF see below. Therefore venting is required.
- C. Incorrect: Exceeding release rates is not authorized as the Release Icon is not associated with this step.
- D. Correct: IAW the discussion section of PNPS 5.7.3.2, failure or unavailability of 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. On the other hand, if an event has occurred and water level cannot be determined or did, in fact, drop below TAF, then the Operator has reason to be concerned that significant amounts of hydrogen may have been generated unless alternate methods of determination indicate otherwise. Alternate methods include sampling, which may involve a long time delay, or deduction based on actual plant status.

This is further reinforced in Discussion Item [2](c) (2) which says that Primary Containment hydrogen cannot be determined to be below 1% when RPV water level has dropped below top of active fuel.

Since it cannot be assumed that [H<sub>2</sub>] is < 1% venting is required per steps G-3 and G-4

of EOP-03. However exceeding release rates is not authorized as the Release Icon is not associated with this step.

Technical Reference(s): PNPS 5.7.3.2 page 5 discussion (Attach if not previously provided)  
section and item [2]

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-05, EO 14 a. b (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		1
	Group #		2
	K/A #	295007	AA2.01
	Importance Rating		4.1

Ability to determine and/or interpret the following as they apply to HIGH REACTOR

PRESSURE: Reactor pressure

Proposed Question: SRO Question # 85

The plant was at rated conditions when a high drywell pressure condition resulted in a reactor scram. Current plant conditions are as follows:

- Torus Bottom pressure: 20 psig, rising slowly
- Drywell temperature: 250 degrees, rising slowly
- Rx Water level: 25 inches, steady
- Rx pressure: 935 steady
- All control rods are inserted
- The MSIVs are open
- Drywell sprays are about to be placed in service

IAW with plant procedures which one of the following RPV pressure control strategies should be implemented?

- A. IAW EOP-01, RPV Control, commence a cooldown by partially opening one main turbine bypass valve and slowly but steadily reduce pressure not to exceed a cooldown rate of 100 degrees/hour.
- B. IAW PNPS 5.3.35.2, Operations Emergency and Transient Response Strategies, fully open one main turbine bypass valve and rapidly lower pressure. Stabilize pressure between 450 to 550 psig.
- C. IAW EOP-01, RPV Control, anticipate an Emergency Depressurization and open all bypass valves.
- D. IAW EOP-17, Emergency RPV Depressurization, open all SRVs and Emergency Depressurize.

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The strategy is to aggressively reduce RPV pressure using available Table M systems when a primary system leak is threatening the containment.
- B. Correct: IAW PNPS 5.3.35.2, Attachment 2, If MHC is available, fully open one main turbine bypass valve using the bypass valve opening jack while maintaining water level within -20 to +45" band. Stabilize pressure 450 to 550 psig to avoid exceeding a 100°F/hour cooldown rate.
- C. Incorrect: Although containment parameters are high plant conditions do not yet currently warrant that the ED be anticipated as drywell sprays have not yet been placed in service.
- D. Incorrect: Although containment parameters are high plant conditions do not yet currently require an Emergency Depress (ED). The criteria for an ED is that the parameter cannot be maintained below its limit, and without first attempting drywell sprays this determination cannot be made.

Technical Reference(s): PNPS 5.3.35.2, Attachment 2, (Attach if not previously provided)  
Sequence 1 and 2

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-02, EO 29 and 30 (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 #		2
	Group #		1
	K/A #	209001	A2.05
	Importance Rating		3.6

Ability to (a) predict the impacts of the following on the LOW PRESSURE CORE SPRAY SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Core spray line break

Proposed Question: SRO Question # 86

Given the following:

- The plant is at rated conditions
- HPCI is out of service for maintenance and the plant is on day 2 of a 14 day LCO per Tech Spec 3.5.C
- Alarm INJECTION HEADER BREAK DETECTION, C903L-C7, annunciates
- The non-licensed operator reports that Core Spray "A" Line break differential pressure, dPIS-1459A, is indicating +5.0 psid

Assuming that conditions do not change, which one of the following

(1) is correct regarding the status of Core Spray "A"

AND

(2) is correct per Technical Specifications?

- A. (1) The "A" Core Spray injection line has failed inside the shroud.  
(2) Plant operation can continue for 7 days. The plant shall then be in cold shutdown within the following 24 hours.
- B. (1) The "A" Core Spray injection line has failed between the vessel wall and the shroud.  
(1) Plant operation can continue for 7 days. The plant shall then be in cold shutdown within the following 24 hours.
- C. (1) The "A" Core Spray injection line has failed inside the shroud.  
(2) An orderly plant shutdown is required and the plant shall be in cold shutdown within the next 24 hours.
- D. (1) The "A" Core Spray injection line has failed between the vessel wall and the shroud.  
(2) An orderly plant shutdown is required and the plant shall be in cold shutdown within the next 24 hours.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: 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 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. Additionally, per TS 3.5.C, from and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable. If these requirements cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.
- B. Incorrect: With Core Spray "A" inop in conjunction with HPCI, the plant must be in cold shutdown within 24 hours.
- C. Incorrect: The line has failed between the vessel wall and the shroud. See "A" above.
- D. Correct: 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 → 0.5 psid, and will therefore cause an alarm in the control room. Per TS 3.5.C, from and after the date that the HPCI system is made or found to be inoperable for any reason, continued reactor operation is permissible only during the succeeding 14 days unless such system is sooner made operable, providing that during such 14 days all active components of the ADS system, the RCIC system, the LPCI system and both core spray systems are operable. If these requirements cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor shall be in the Cold Shutdown Condition within 24 hours.

Technical Reference(s): Core Spray reference text, page 18 (Attach if not previously provided)  
TS 3.5.C,

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-02, EO 2g and 15 (As available)

Question Source: Bank # WTS 1190  
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 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	203000	A2.12
	Importance Rating		2.7

A2.12 - 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: Pump runout  
Proposed Question: SRO Question # 87

EOP-26, RPV Flooding, Failure to Scram is being executed. The following conditions exist.

- All SRVs are open
- The Minimum Steam Cooling Pressure has been established
- RPV pressure is being maintained just above the required Minimum Steam Cooling Pressure via RHR injection
- RHR pumps "A" and "C" are injecting
- MO-1001-28A, LPCI Injection Valve #1, is throttled
- Total RHR injection flow is 6000 gpm
- Both Core Spray pumps are available but in standby
- No other sources of injection are available

Then, RHR Pump "A" trips.

Which one of the following is correct?

Pressure will fall below the Minimum Steam Cooling Pressure (MSCP) and .....

- A. can be re-established by throttling open MO-1001-28A, LPCI Injection Valve #1.
- B. cannot be re-established using RHR. Start Core Spray pumps as required to re-establish the MSCP.
- C. cannot be re-established using RHR. Core Spray injection is prohibited. Exit EOP-26 and enter SAGs.
- D. can be re-established by opening additional vent paths IAW EOP-26, Table P, Alternate RPV Depressurization Systems which will lower the required MSCP.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The single RHR pump cannot inject at 6000 gpm without going into runout. Even if the candidate decides that exceeding the limit is necessary, the runout condition will result in very low discharge pressures preventing the re-establishment of the MSCP.

B. Correct: Prior to the pump trip each pump was injecting at 3000 gpm in order to maintain the required MSCP (240 psig w/4 SRVs open). One RHR pump cannot inject 6000 gpm without going into runout as the runout limit is 5600 gpm. Even if the candidate decides that exceeding the limit is necessary, the runout condition will result in very low discharge pressures preventing the re-establishment of the MSCP.

Step F-11 of EOP-26 directs the operator to establish the MSCP using outside the shroud systems and if necessary Table X systems. Although not preferred, Core Spray injection is required in order to avoid core damage.

C. Incorrect: Core Spray injection is allowed if required. SAGs are entered when core damage is indicated. Plausible in that Core Spray injection is avoided if at all possible during ATWS conditions as the water is not pre-heated and may "flush out" any boron that has been injected.

D. Incorrect: Injecting with Core Spray is the required action. Plausible in that alternate vent systems will be used if the MSCP cannot be established depending upon SRV status. However the other criteria for using alternate vent systems is if 4 SRVs cannot be opened.

Technical Reference(s): PNPS 2.2.19 Precaution # 2 (Attach if not previously provided)  
EOP-26, step F-11, Tables Y and X

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-04-08, EO13 (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:



Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	215004	2.4.30
	Importance Rating		4.1

Emergency Procedures / Plan; Knowledge of events related to system operation / status that must be reported to internal organizations or external agencies, such as the state, the NRC, or the transmission system operator. (Source Range Monitor)

Proposed Question: SRO Question # 88

The plant is refueling with a core alteration in progress when the following sequence occurs:

- Time: 00:00
  - SRM counts begin to rise
  - Sustained SRM Period alarms are received on two channels
  - SRM counts continue to rise
  - Manual Reactor scram inserted due to rising counts on multiple channels
  - SRM counts lower to original values
  - Refuel Floor Vent Exhaust monitors begin to rise
- Time: 00:20
  - Refuel Floor Vent Exhaust Hi Alarms are received on all 4 channels.
  - Main Stack Low Range Rad monitors RM-1705-18A & B begin to rise
- Time: 00:40
  - Main Stack Low Range Rad monitors RM-1705-18A & B indicate 4E+5 cps and rising slowly
- Time 00:55
  - Main Stack Low Range Rad monitors RM-1705-18A & B indicate 8E+5 CPS and are stable

IAW EPIP-100 Emergency Classification And Notification, which one of the following describes ALL the reports to be made to the state and local agencies and the bases for those reports?

- A. Initial Notification of an Alert based on exceeding an Alert EAL at 00:20
- B. Initial Notification of an Unusual Event (UE) based on exceeding an UE EAL at 00:00  
Follow-up Notification due to having a release in progress at 00:40
- C. Initial Notification of an Unusual Event (UE) based on exceeding an UE EAL at 00:00  
Initial Notification of an Alert based on exceeding an Alert EAL at 00:55
- D. Initial Notification of an Unusual Event (UE) based on exceeding an UE EAL at 00:00  
Initial Notification of an Alert based on exceeding an Alert EAL at 00:20

Follow-up Notification based on having a release in progress at 00:55

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: A UE was exceeded at time 00:00 based on EAL CU5.1, Unplanned sustained positive period observed on nuclear instrumentation. Also a Follow-up Notification is required due to meeting the criteria of a Release in progress at time 00:55. Plausible if the Candidate does not recognize the UE or understand what a release in progress is.
- B. Incorrect: An Alert was exceeded at time 00:20 when Refuel Floor Vent Exhaust Hi Alarms were received on all 4 channels. Also the criteria for declaring a release in progress was not met until 00:55. Time 00:40 is plausible for this since the level for the Alert Rad Release EAL was exceeded at this time. However that level must be present for 15 minutes.
- C. Incorrect: The Alert was first exceeded at time 00:20 which would require an initial notification. Although an Alert EAL was also exceeded at time 00:55 due to Main Stack monitors being above  $4E+5$  for 15 minutes, an Initial Notification should have been sent out based on exceeding the Alert EAL at time 00:20.
- D. Correct: An Unusual Event is required based on EAL CU5.1, unplanned sustained positive period observed on nuclear instrumentation. The initial SRM response indicates an unplanned criticality and a UE is required.

When Refuel Floor Vent Exhaust Hi Alarms were received on all 4 channels at time 00:20 an Alert is required based on EAL AA2.1, Damage to irradiated fuel OR loss of water level (uncovering irradiated fuel outside the RPV) that causes a valid high alarm on any of the following radiation monitors (Panel C910/C911):

- New Fuel Vault (RIS-1815-3D)
- Refuel Floor Shield Plug Area (RIS-1815-3E)
- Spent Fuel Pool Area (RIS-1815-3F)
- Refuel Floor Vent Exhaust (RIS-1705-8A-D)

Since the Refuel Floor Vent Exhaust monitors began to rise at the same time of the inadvertent criticality it must be assumed that damage to irradiated fuel has occurred.

At time 00:55 the criteria for declaring a release in progress was met. The definition of a release in progress is any release of radioactivity which meets any EAL of Subcategory A1, Offsite Rad Conditions or Involves an actual or suspected Turbine Building or unmonitored release which is associated with the emergency event.

At time 00:55 Subcategory A1 EAL AA1.1, Any valid gaseous monitor reading  $>$  Table A-1 column "Alert" for 15 min was exceeded. At time 00:55, the Main Stack Rad monitors have been above the Alert Value of  $4E+5$ CPS for 15 minutes.

Since an upgrade to a higher EAL is not warranted the information regarding the Release in Status is transmitted via a Follow-Up notification.

Technical Reference(s): Cold EAL Chart (Attach if not previously provided)  
EPIP-100, definition [11] on page  
7

Proposed References to be provided to applicants during examination: EAL Chart

Learning Objective: O-RO-07-02-01, EO 2 (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	215003	2.2.38
	Importance Rating		4.5

Equipment Control: Knowledge of conditions and limitations in the facility license. (IRM)

Proposed Question: SRO Question # 89

The plant is refueling with the following plant conditions:

- All control rods are inserted
- Fuel is being loaded in the vessel

Then, I&C reports that the IRM High Rod Block trip for all IRMs is set at 110/125<sup>ths</sup> of scale.

Which one of the following is correct?

Core alterations ...

- can continue but one IRM must be placed in the tripped condition within one hour.
- must be immediately suspended until one IRM is placed in the trip condition.
- can continue and a Tracking LCO must be generated.
- must be immediately suspended.

Proposed Answer: C

Explanation (Optional):

- Incorrect: The IRM function is not required. Plausible in that this would be the required Tech Spec action if they were required.
- Incorrect: There is no requirement to suspend core alterations until one IRM is placed in trip.
- Correct: Per Table 3.2.C-2 of the FSAR, the IRM Upscale Rod Block setpoint must be < 108/125 of full scale. Therefore the IRM rod block function is inoperable for all IRMs. However, the IRM function is not required in refuel if the head is removed and all rods are inserted per Note 6 of FSAR Table 3.2.C-1. Since they are not required, a Tracking

LCO is required.

- D. Incorrect: There is no requirement to suspend core alterations. Plausible in that the IRM function is required in refuel under certain conditions.

Technical Reference(s): FSAR Table 3.2.C-1, and (Attach if not previously provided)  
associated notes  
PNPS 1.3.34.2, Definition [12]  
FSAR Table 3.2.C-2

Proposed References to be provided to applicants during examination: FSAR Table 3.2.C-1, and associated notes

Learning Objective: O-RO-02-07-02, EO 16 (As available)

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
55.43 7

Fuel handling facilities and procedures.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		1
	K/A #	264000	2.2.40
	Importance Rating		4.7

Equipment Control: Ability to apply technical specifications for a system (EDGs).

Proposed Question: SRO Question # 90

PNPS is at rated conditions with an overhaul of the "B" EDG in progress. The plant is on day 3 of a 14 day LCO due to the diesel maintenance.

Then, the breaker for MO-1001-36B, RHR Loop B Torus Cooling Valve is determined to be inoperable and the valve cannot be opened from the control room.

MO-1001-34B, Torus Cooling/Spray Block Valve, is then verified closed and is disabled in the closed position.

All other equipment is operable.

If conditions do not change, the plant must be placed in COLD SHUTDOWN conditions within:

- A. 1 day
- B. 4 days
- C. 8 days
- D. 12 days

Proposed Answer: A

Explanation (Optional):

- A. Correct – The breaker for RHR valve 36B being INOP renders the "B" Torus Cooling and the "B" Containment Spray subsystems INOP. This in of itself is a 7 day LCO (TS 3.5.B.1 and 2). However the EDG outage requires that all low pressure core and containment cooling systems be operable IAW TS 3.5.F.1 while the EDG is not available. If this requirement is not met, the reactor must be placed in cold shutdown in 24 hours (1 day).
- B. Incorrect – The plant must be in cold SD within 24 hours. 4 days is obtained if the

candidate applies the 1<sup>st</sup> part of LCO 3.5.F that addresses an LCO time of 72 hours with the diesel OOS and then adds 24 hours to achieve Cold S/D.

- C. Incorrect – The plant must be in cold SD within 24 hours. 8 days is obtained if the candidate does not understand the relationship between the Torus Cooling and the Containment Spray subsystems and the Inop EDG. In this case the candidate would utilize the 7 day LCO of TS 3.5.B.1 and 2 and add 24 hours to achieve cold S/D.
- D. Incorrect – The plant must be in cold SD within 24 hours. The 12 days is obtained if the candidate only considers MO-1001-36B as a PCIS valve. In that case then the TS action of disabling the MO-1001-34B would have restored the Primary Containment and the limiting LCO would then revert back to the original EDG LCO time. Since there is 11 days left on the original EDG LCO time and 24 hours is allowed to obtain cold S/D the candidate could arrive at a time frame of 12 days.

Technical Reference(s): TS sections TS 3.5.B.1 and 2 and (Attach if not previously provided) section 3.5.F

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-06, EO 30 (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  
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 #		2
	K/A #	256000	A2.02
	Importance Rating		2.9

A2.02 - Ability to (a) predict the impacts of the following on the REACTOR CONDENSATE SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Valve closures  
Proposed Question: SRO Question # 91

The plant is at rated conditions with all components operable.

Then TBCCW non-essential block supply MO-4127 fails closed and cannot be re-opened.

Which one of the following is correct regarding:

- (1) The impact on the Condensate system AND
  - (2) The required action?
- A. (1) Condensate pumps trip on low cooling water flow after 60 seconds.  
(2) Enter PNPS 2.4.41, Loss of TBCCW, scram the reactor, secure all condensate and reactor feed pumps and close the MSIVs.
  - B. (1) Condensate pumps lose cooling water flow and begin to overheat.  
(2) Enter PNPS 2.4.41, Loss of TBCCW, scram the reactor and secure all condensate and reactor feed pumps and close the MSIVs.
  - C. (1) Condensate pumps lose cooling water flow and begin to overheat.  
(2) Enter PNPS 2.1.14, Station Power changes and insert the RPR array. Secure any condensate pump whose bearing temperatures exceed the limits of ARP C2R-B7, Omniguard Alarm.
  - D. (1) Condensate pumps trip on low cooling water flow after 60 seconds.  
(2) Enter PNPS 2.4.49, Feedwater Malfunctions, scram the reactor, trip all reactor feed pumps when the condensate pumps trip. Close the MSIVs when vacuum can no longer be maintained.

Proposed Answer: B



Explanation (Optional):

- A. Incorrect: The condensate pumps will not automatically trip on low flow. Plausible in that the reactor feed pumps did until very recently.
- B. Correct: The condensate pumps will lose cooling water flow. PNPS 2.4.41, Loss of TBCCW would be applicable procedure. The subsequent actions of this procedure are to scram, secure all feed and condensate and close the MSIVs.
- C. Incorrect: The appropriate action would be to respond IAW 2.4.41. Plausible in that reducing power would allow removing a condensate pump from service and the referenced alarm does provide safe operating temperature limits for the condensate pumps.
- D. Incorrect: The condensate pumps will not automatically trip. Also the MSIVs are closed when the last condensate pump is secured. Closing the MSIVs when vacuum begins to degrade is plausible because condensate flow cools the Gland Seal and SJAE condensers.

Technical Reference(s): PNPS 2.4.41, Subsequent action. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-02-03 TO, RESPOND TO (As available)  
LOSS OF TBCCW PUMPS.

Question Source: Bank #  
Modified Bank #  
New X

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41  
55.43 5

Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		2
	Group #		2
	K/A #	202002	2.4.11
	Importance Rating		4.2

Emergency Procedures / Plan: Knowledge of abnormal condition procedures. (Recirculation Flow Control)

Proposed Question: SRO Question # 92

The Reactor is operating at rated conditions when a FWLC malfunction results in a dual Recirc pump runback to minimum speed. Current plant conditions are:

- Reactor power: 78%
- Core flow: 33 MLbm/hr
- Both Recirc Pump scoop tubes have been locked

Which one of the following is correct for the above conditions?

- Enter PNPS 2.4.165, Reactor Core Instability, and if Decay Ratios are acceptable operation can continue at the current power level while establishing local control of the Recirc MG Sets.
- IAW PNPS 2.1.14, Station Power Changes, control rods must be inserted in reverse order of the pull sheet to reduce power to <73%.
- IAW PNPS 2.1.14, Station Power Changes, the RPR array must be inserted to reduce power to < 73%.
- Enter PNPS 2.1.6, Reactor Scram, and insert a manual reactor scram.

Proposed Answer: C

Explanation (Optional):

- Incorrect: Power needs to be reduced. Plausible in that the reactor is also in the Buffer Zone. PNPS 2.4.165, subsequent action [5] directs that core decay ratios be checked against their limits. If acceptable, then operation can continue in the buffer zone. However being above MELLA requires the power reduction.
- Incorrect: Use of the RPR array is specified. Plausible in that if power were below the 60% load line, rods would be inserted using the reverse order of the pull sheet.

C. Correct: The plant is operating above the MELLA line. PNPS 2.4.20, subsequent action [2] states: "IF Reactor power is above MELLLA line, THEN REDUCE power in accordance with PNPS 2.1.14 Section 7.9 until below the MELLLA line."

Lowering power to < 73% will restore operation to below the MELLA line. PNPS 2.1.14 section 7.9 directs use of the RPR since power is above the 60% load line.

D. Incorrect: The required action is to insert the RPR array. Plausible in that the plant is in the Operation Prohibited Region (above the MELLA line).

Technical Reference(s): PNPS 2.4.165, subsequent actions (Attach if not previously provided)  
PNPS 2.4.20, subsequent actions  
PNPS 2.1.14, section 7.9

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

Learning Objective: O-RO-03-03-09(02), TO RESPOND (As available)  
TO A RECIRC SYSTEM SPEED OR  
FLOW CONTROL MALFUNCTION

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 #		2
	Group #		2
	K/A #	204000	2.4.12
	Importance Rating		4.3

Knowledge of general operating crew responsibilities during emergency operations. (RWCU)  
Proposed Question: SRO Question # 93

The plant is at 30% power when the NRC contacts PNPS via the ENS line. The NRC reports that the FBI has determined that a credible threat exists specifically for Pilgrim station.

Details regarding the timing of the threat are not currently known.

Which one of the following is required?

- A. Enter PNPS 2.1.14, Station Power Changes and commence a power reduction. No other action is required unless an attack is imminent.
- B. Enter PNPS 2.1.14, Station Power Changes, commence a power reduction and isolate RWCU.
- C. Enter PNPS 2.1.6 Reactor Scram, scram the reactor and isolate the Main Steam Lines.
- D. Enter PNPS 2.1.6 Reactor Scram, scram the reactor and isolate RWCU.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: Because the threat is both credible and valid, a reactor scram is warranted. Additionally isolating RWCU is required.
- B. Incorrect: Because the threat is both credible and valid, a reactor scram is warranted.
- C. Incorrect: The MSIVs are not isolated in order to preserve the main condenser as a heat sink.
- D. Correct: The threat is credible because it came from the FBI. The threat is also classified as a Valid credible threat because it came from the NRC via the ENS line. Attachment 1 of PNPS 5.3.14, Step [3] then requires a scram and that RWCU be isolated.

Technical Reference(s): PNPS 5.3.14, Attachment 1, definitions and section 4.0. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # PNPS 9453 Modified stem and distractors to include procedure.  
Modified Bank #  
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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.1.25
	Importance Rating		4.2

Ability to interpret reference materials, such as graphs, curves, tables, etc.

Proposed Question: SRO Question # 94

The plant is in cold shutdown with RHR aligned for Shutdown Cooling (SDC). Additional information is as follows:

- The plant was shutdown 3 days ago.
- Prior to the shutdown PNPS had been on line for 185 days.
- A loss of off-site power has just occurred.
- Immediately prior to the loss of off-site power, plant parameters were as follows:
  - RPV level 30 inches
  - RPV moderator temperature 138 degrees
  - Drywell is open to allow repair of a leaking head vent valve

Assuming no operator actions are taken, the 1<sup>st</sup> Tech Spec violation will occur within \_\_\_\_ (1) \_\_\_\_.

AND

If a Tech Spec violation DOES occur, and the Shift Manager has determined that design isolation signals must be defeated in order to mitigate the event, NRC notification is required within \_\_\_\_ (2) \_\_\_\_.

Notes: Do not consider any required NRC notifications required by the Emergency Plan. References are provided.

- A. (1) 1.5 hours  
(2) 1 hour
- B. (1) 6.7 hours  
(2) 1 hour
- C. (1) 1.5 hours  
(2) 4 hours
- D. (1) 6.7 hours  
(2) 4 hours

Proposed Answer: A

Explanation (Optional):

- A. Correct: Per Attachment 5 of 2.4.25, and vessel level at 30", and 3 days after shutdown, heatup rate is 49.8 degrees/hr. A Tech Spec violation will occur when moderator temperature reaches 212 degrees since the drywell is open.  $(212-138)/49.8 = 1.49$  hrs.

As discussed in PNPS 2.4.25, page 9 and PNPS 1.3.12, Attachment 12, IF corrective action to recover from this situation involves inhibition of design isolation signals by jumpers/lifted leads/booted contacts or use of the Alternate Shutdown Panel, THEN notification to the NRC of the Station's intended actions is required. This is a 1-hour report as required by 10CFR50.54X.

- B. Incorrect: The 1<sup>st</sup> Tech Spec Violation will occur when temperature exceeds 212 degrees. Plausible in 6.7 hours is the amount of time that it will take for level to lower to TAF. If the candidate considers the cold SD safety limit of 12 inches above the TAF, the candidate may choose this response.
- C. Incorrect: The NRC must be notified within 1hour. Plausible if the candidate does not recognize this as a 50.54X event and applies the reporting requirement for the initiation of a plant shutdown required by Tech Specs. (4 hour report by 50.72(b)(2)(i))
- D. Incorrect: Incorrect the 1<sup>st</sup> tech spec violation will occur when moderator temperature reaches 212 degrees. Additionally, a 1 hour report is required.

Technical Reference(s): PNPS 2.4.25, LOSS OF SHUTDOWN COOLING, Attachment 5 and page 9  
PNPS 1.3.12, Attachment 2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: PNPS 2.4.25, LOSS OF SHUTDOWN COOLING, Attachment 5  
PNPS 1.3.12, Attachment 2

Learning Objective: (As available)

Question Source: Bank # Modified Bank # LOR Bank #55 Modified to include reporting requirements for 50.54.X

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 #		
	K/A #		G2.2.21
	Importance Rating		4.1

Knowledge of pre- and post-maintenance operability requirements.

Proposed Question: SRO Question # 95

The plant is at rated conditions. HPCI has been removed from service for corrective maintenance. An LCO was generated IAW PNPS 1.3.34.2, Limiting Conditions for Operation Log.

Which of the following statements is/are correct regarding the closing of the LCO at the conclusion of the maintenance?

- # 1 All condition reports documenting discrepancies identified during the maintenance must be closed.
- # 2 Operability Evaluations must be completed for all condition reports documenting discrepancies identified during the maintenance.
- # 3 Components that reposition or change state during an automatic initiation need not be in a Standby lineup provided all required power supplies are available to support an auto initiation.

- A. 1 only
- B. 2 only
- C. 2 and 3
- D. 1 and 3

Proposed Answer: B

Explanation (Optional):

- A. Incorrect: The condition report need not be closed. Only an Operability Eval must be completed.

- B. Correct: IAW 1.3.34.2, section 6.5, step [2] (g), if the LCO condition involves a Condition Report documenting a degraded or nonconforming condition, then an approved Operability Evaluation must exist prior to the clearance of the LCO.
- C. Incorrect: Statement 3 is not correct. Per PNPS 1.3.34.2 section 6.5 and Attachment 5, item [8], the system is required to be restored to a standby lineup or as determined by the SM to be acceptable for the current mode. Since the plant is at rated conditions, HPCI would be placed in a standby lineup.
- D. Incorrect: The condition report need not be closed. Only an Operability Eval must be completed. Also HPCI is required to be in a standby lineup.

Technical Reference(s): PNPS 1.3.34.2 section 6.5 and Attachment 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01, 28a (As available)

Question Source: Bank #  
 Modified Bank #  
 New X

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 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.3.13
	Importance Rating		3.8

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 # 96

A drywell entry is required. The Radiation Protection Manager has given approval for the entry.

Select the choice below that correctly completes the following regarding a containment entry IAW PNPS 1.4.12, Primary Containment Entry.

In order for the Shift Manager to authorize the containment entry WITHOUT obtaining any ADDITIONAL approvals:

- Reactor power level must be less than \_\_\_ (1) \_\_\_%.
- The containment atmosphere \_\_\_ (2) \_\_\_ required to be de-inerted prior to the entry.

- A. (1) 5  
(2) is
- B. (1) 5  
(2) is NOT
- C. (1) 10  
(2) is
- D. (1) 10  
(2) is NOT

Proposed Answer: A

Explanation (Optional):

- A. Correct: When the containment is de-inerted and Reactor power level is less than 5% CTP, Drywell entry may be authorized by the Shift Manager provided that approval has been obtained from the Radiation Protection Manager (RPM).

- B. Incorrect: The containment must be de-inerted. Plausible in that there is a process for allowing entry into an inerted containment but that requires additional approvals.
- C. Incorrect: Power must be < 5%. Plausible in that there is a process for allowing entry at higher powers but that requires additional approvals.
- D. Incorrect: Power must be < 5% and the containment must be de-inerted. Plausible in that there is a process if these conditions cannot be met during emergencies but additional approvals are required.

Technical Reference(s): PNPS 1.4.12, section 4.1 and also (Attach if not previously provided) Attachment 1 pages 2 and 3.

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-06-01 (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 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 #		
	K/A #		G2.4.35
	Importance Rating		4.0

Knowledge of local auxiliary operator tasks during emergency and the resultant operational effects.

SRO Level Justification: IAW PNPS 2.1.7, VESSEL HEATUP AND COOLDOWN the CRS is required to review the OPER-07 data after each sheet of readings.

Proposed Question: SRO Question # 97

The plant was operating in single loop due to corrective maintenance on Recirc MG set "A". The following then occurs:

- Recirc MG Set "B" trips
- The reactor is manually scrammed
- RWCU is placed back in service following the RWCU isolation
- A reactor plant cooldown is commenced using the main turbine bypass valves

Additional information is as follows:

- RPV level is +30 inches on the Narrow Range indicators
- OPER-07 Cooldown Data is as shown in the table below
- At time 00:60, the fault on Recirc MG Set "B" is repaired

Date	TR-260-151A & B Panel C904		PR/FR-640-27 Panel C905	TR-263-104 Panel C921	TR-263-104 Panel C921		
	Recirc/RHR Loop Suction Temp		Rx Wide Range Pressure	(Pt. 2)	(Pt. 2)	(Pt. 4)	(Pt. 9)
Time	Loop A	Loop B		Vessel Shell Adj To Flange (RXX022)	Line Vessel Drain (RXX024)	Vessel Below Water Level (RXX024)	Vessel Bottom Head (RXX030)
00:00	510	500	900	531	505	528	506
00:15	499	492	800	518	468	516	467
00:30	472	468	700	503	428	500	430
00:45	450	448	600	486	365	485	370
00:60	425	426	500	467	298	464	308

IAW PNPS 2.4.24, Reactor Vessel Cold Water Stratification, and the information provided, which one of the following is actions is authorized?

Directing the ...

- A. RO to start Recirc MG Set "B".
- B. RO to raise RPV level to > + 60 inches as indicated on LI-263-101.
- C. NLO to increase RWCU system flow to the filter demineralizer stretch rating.
- D. NLO to throttle close MO-1201-85, RWCU Recirculation Loop Suction Valve.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: There is a greater than 145 °F between the bottom head drain temperature and saturation temperature. Tech Spec 3.6.A, and OPER 19 preclude starting the recirc pump. Plausible in that this would be an acceptable action to reduce vessel stratification if recirc pump start limitations were met.
- B. Incorrect: Although the vessel is stratifying, RPV level is not raised because the main condenser is being used as a heat sink. This action is only taken if a heat sink is not available in order to increase natural circulation.
- C. Incorrect: This action is not directed by PNPS 2.4.24. Plausible in the procedure does direct increasing RWCU flow from the bottom head drain.
- D. Correct: The vessel is stratifying. Throttling closed on the RWCU-85 valve will result in increasing flow from the bottom head region.

Technical Reference(s): PNPS 2.4.24, Attachment 1, sheet (Attach if not previously provided)  
 1.  
 Tech Spec 3.6.A and OPER 19  
 start limitations.

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

Learning Objective: O-RO-02-06-05 , EO 14a (As available)

Question Source: Bank #  
 Modified Bank #  
 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 #		
	K/A #		G2.2.23
	Importance Rating		4.6

Ability to track Technical Specification limiting conditions for operations.

K/A Justification: In order to track Tech Spec LCOs it is necessary to know when the clock started for the LCO.

Proposed Question: SRO Question # 98

Given the following sequence:

- At time 00:00: RBCCW Pump "A" is started and pump "B" secured
- At time 00:30: An operator reports that the RBCCW Pump "A" seal is leaking
- At time 01:00: The seal leak is quantified as 40 drops/minute
- At time 01:20: The Shift Manger contacts the system engineer to assist in an Operability Determination
- At time 03:00: The system engineer reports that the total RBCCW loop "A" leakage now exceeds the 240 drops per minute limit and that the RBCCW loop will not meet its required mission time of 30 days.
- At time 03:15: The Shift Manager reviews and concurs with the system engineer's calculations and declares the "A" RBCCW loop inoperable.

In accordance with EN-OP-104, Operability Determination Process, the time clock (LCO time) for the associated Limiting Condition of Operation, Action Statement started at:

- A. 00:00 since the seal most likely began leaking upon initial start.
- B. 00:30 since this was the time of discovery.
- C. 01:00 since this is the time the leakage was quantified and when the limit of 240 gpm was known to have been exceeded.
- D. 03:15 since this was the time the shift manager declared the loop inoperable.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: Per EN-OP-104, the clock time starts when the SM makes the operability determination.



- B. Incorrect: Per EN-OP-104, the clock time starts when the SM makes the operability determination.
- C. Incorrect: Per EN-OP-104, the clock time starts when the SM makes the operability determination.
- D. Correct: Per EN-OP-104, and regarding Operability Determination Process Times the Time of Determination is the moment the Shift Manager (SM) decides on the status of the condition. The time clock for a Limiting Condition of Operation, Action Statement or reportable condition starts at the time of determination.

Technical Reference(s): EN-OP-104, Operability Determination Process, page 15 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-06-01-06, TO, 299-03-05-005 (As available)  
 CHECK SYSTEMS GOVERNED BY  
 TECH SPECS. TO ENSURE  
 OPERABILITY REQUIREMENTS  
 ARE MET.

Question Source: Bank #  
 Modified Bank # PNPS 11479 Modified to provide an actual example  
 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 2

Facility operating limitations in the technical specifications and their bases.

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.3.11
	Importance Rating		4.3

Ability to control radiation releases.

Proposed Question: SRO Question # 99

A main steam leak in the turbine building has occurred that cannot be isolated. EOP-05, Rad Release Control, has been entered.

IAW EOP-05, an Emergency Depressurization is required before ...

An EAL chart is provided for your use.

- A. Main Stack High Range Effluent Monitor RI-1001 has exceeded 2 R/Hr for 15 minutes.
- B. Main Stack Low Range PRMs 1705-18A and B have exceeded 400,000 cps for 15 minutes.
- C. The dose at the site boundary exceeds 100 mRem whole body as indicated by dose assessment.
- D. The dose at the site boundary exceeds 1000 mRem whole body as indicated by dose assessment.

Proposed Answer: D

Explanation (Optional):

- A. Incorrect: EOP-05 requires that an Emergency Depressurization be conducted before the General Emergency Offsite Release limits are exceeded. A Main Stack High Range monitor of 2 R/Hr for 15 minutes is the threshold for a Site Area Emergency.
- B. Incorrect: Main Stack Low Range PRMs reading 400,000 cps for 15 minutes is the threshold for an Alert. Plausible in that there is a specified EOP-05 action to be taken when the offsite release exceeds the limits of an Alert EAL.
- C. Incorrect: A dose assessment that indicates that the dose rate at the site boundary will exceed 100 mRem/hr is the threshold for a Site Area Emergency.
- D. Correct: EOP-05 requires that an Emergency Depressurization be conducted before the General Emergency Offsite Release limits are exceeded. A General Emergency

EAL has been exceeded when dose projections indicate that the dose rate at the site boundary will exceed 1000 mRem/hr.

Technical Reference(s): EOP-05, step (Attach if not previously provided)  
EAL Chart

Proposed References to be provided to applicants during examination: HOT EAL Chart

Learning Objective: O-RO-03-04-07, EO 7a (As available)

Question Source: Bank #  
Modified Bank #  
New X

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:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		3
	Group #		
	K/A #		G2.4.34
	Importance Rating		4.1

Knowledge of RO tasks performed outside the main control room during an emergency and the resultant operational effects.

K/A Justification: IAW PNPS 2.4.143, operation of the SRVs from their ASPs can only be assigned to licensed ROs.

Proposed Question: SRO Question # 100

The Main Control Room has just been abandoned due to a fire.

What action is required to be directed regarding the Alternate Shutdown Panels for the SRVs?

- A. Place all SRVs in Close at their ASPs within 15 minutes to terminate any spurious SRV operation.
- B. Place all SRVs in Close at their ASPs within 24 minutes to terminate any spurious SRV operation.
- C. Take local control of at least one SRV within 24 minutes and control pressure between 900 and 1050 psig to prevent excessive SRV cycling on high pressure.
- D. Take local control of at least one SRV within 15 minutes and control pressure between 900 and 1050 psig to prevent excessive SRV cycling on high pressure.

Proposed Answer: A

Explanation (Optional):

- A. Correct: IAW the Caution on page 9 of PNPS 2.4.143, local control of all SRVs must be established within 15 minutes to prevent spurious SRV operation.
- B. Incorrect: The required time frame is 15 minutes. The 24 minute time frame is the time frame for establishing injection flow.
- C. Incorrect: There is no directed action to establish pressure control within a specified time frame. However this is the specified band when pressure control is first established.

D. Incorrect: There is no directed action to establish pressure control within a specified time frame.

Technical Reference(s): 2.4.143 page 9

(Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-03-26(01) TO, SUPERVISE (As available)  
CREW DURING SHUTDOWN  
OUTSIDE THE CONTROL ROOM

Question Source: Bank #  
Modified Bank #  
New X

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:

U.S.N.R.C. Site-Specific Written Examination  
Pilgrim  
Senior Reactor Operator

2014  
KEY

51.	A	B	C	<input checked="" type="radio"/>	76.	<input checked="" type="radio"/>	B	C	D
52.	<input checked="" type="radio"/>	B	C	D	77.	A	B	C	<input checked="" type="radio"/>
53.	<input checked="" type="radio"/>	B	C	D	78.	A	<input checked="" type="radio"/>	C	D
54.	A	<input checked="" type="radio"/>	C	D	79.	A	B	<input checked="" type="radio"/>	D
55.	<input checked="" type="radio"/>		C	D	80.	A	B	<input checked="" type="radio"/>	D
56.	A	B	<input checked="" type="radio"/>	D	81.	A	<input checked="" type="radio"/>	C	D
57.	A	B	<input checked="" type="radio"/>	D	82.	A	B	<input checked="" type="radio"/>	D
58.	A	B	<input checked="" type="radio"/>	D	83.	A	B	<input checked="" type="radio"/>	D
59.	A	B	<input checked="" type="radio"/>	D	84.	A	B	C	<input checked="" type="radio"/>
60.	A	B	C	<input checked="" type="radio"/>	85.	A	<input checked="" type="radio"/>	C	D
61.	A	<input checked="" type="radio"/>	C	D	86.	A	B	C	<input checked="" type="radio"/>
62.	A	B	C	<input checked="" type="radio"/>	87.	A	<input checked="" type="radio"/>	C	D
63.	A	B	<input checked="" type="radio"/>	D	88.	A	B	C	<input checked="" type="radio"/>
64.	A	B	<input checked="" type="radio"/>	D	89.	A	B	<input checked="" type="radio"/>	D
65.	A	B	C	<input checked="" type="radio"/>	90.	<input checked="" type="radio"/>	B	C	D
66.	<input checked="" type="radio"/>	B	C	D	91.	A	<input checked="" type="radio"/>	C	D
67.	A	<input checked="" type="radio"/>	C	D	92.	A	B	<input checked="" type="radio"/>	D
68.	A	<input checked="" type="radio"/>	C	D	93.	A	B	C	<input checked="" type="radio"/>
69.	A	B	C	<input checked="" type="radio"/>	94.	<input checked="" type="radio"/>	B	C	D
70.	A	B	<input checked="" type="radio"/>	D	95.	A	<input checked="" type="radio"/>	C	D
71.	A	B	C	<input checked="" type="radio"/>	96.	<input checked="" type="radio"/>	B	C	D
72.	A	<input checked="" type="radio"/>	C	D	97.	A	B	C	<input checked="" type="radio"/>
73.	A	B	<input checked="" type="radio"/>	D	98.	A	B	C	<input checked="" type="radio"/>
74.	A	B	<input checked="" type="radio"/>	D	99.	A	B	C	<input checked="" type="radio"/>
75.	A	B	C	<input checked="" type="radio"/>	100.	<input checked="" type="radio"/>	B	C	D

KEY

U.S.N.R.C. Site-Specific Written Examination  
Pilgrim  
Senior Reactor Operator

2014

1.	A	B	C	<input checked="" type="radio"/>
2.	A	B	<input checked="" type="radio"/>	D
3.	A	<input checked="" type="radio"/>	C	D
4.	<input checked="" type="radio"/>	B	C	D
5.	A	<input checked="" type="radio"/>	C	D
6.	A	B	<input checked="" type="radio"/>	D
7.	A	B	<input checked="" type="radio"/>	D
8.	A	<input checked="" type="radio"/>	C	D
9.	A	<input checked="" type="radio"/>	C	D
10.	A	B	<input checked="" type="radio"/>	D
11.	A	<input checked="" type="radio"/>	C	D
12.	<input checked="" type="radio"/>	B	C	D
13.	A	<input checked="" type="radio"/>	C	D
14.	A	<input checked="" type="radio"/>	C	D
15.	A	<input checked="" type="radio"/>	C	D
16.	A	<input checked="" type="radio"/>	C	D
17.	<input checked="" type="radio"/>	B	C	D
18.	<input checked="" type="radio"/>	B	C	D
19.	<input checked="" type="radio"/>	B	C	D
20.	A	B	C	<input checked="" type="radio"/>
21.	<input checked="" type="radio"/>	B	C	D
22.	<input checked="" type="radio"/>	B	C	D
23.	<input checked="" type="radio"/>	B	C	D
24.	A	<input checked="" type="radio"/>	C	D
25.	A	B	C	<input checked="" type="radio"/>
26.	<input checked="" type="radio"/>	B	C	D
27.	A	B	<input checked="" type="radio"/>	D
28.	A	<input checked="" type="radio"/>	C	D
29.	A	B	<input checked="" type="radio"/>	D
30.	A	B	C	<input checked="" type="radio"/>
31.	<input checked="" type="radio"/>	B	C	D
32.	A	B	C	<input checked="" type="radio"/>
33.	<input checked="" type="radio"/>	B	C	D
34.	A	<input checked="" type="radio"/>	C	D
35.	A	B	<input checked="" type="radio"/>	D
36.	<input checked="" type="radio"/>	B	C	D
37.	A	<input checked="" type="radio"/>	C	D
38.	A	B	<input checked="" type="radio"/>	D
39.	A	B	<input checked="" type="radio"/>	D
40.	A	B	C	<input checked="" type="radio"/>
41.	A	<input checked="" type="radio"/>	C	D
42.	A	B	<input checked="" type="radio"/>	D
43.	A	<input checked="" type="radio"/>	C	D
44.	A	<input checked="" type="radio"/>	C	D
45.	A	<input checked="" type="radio"/>	C	D
46.	A	B	<input checked="" type="radio"/>	D
47.	A	B	<input checked="" type="radio"/>	D
48.	<input checked="" type="radio"/>	B	C	D
49.	A	B	C	<input checked="" type="radio"/>
50.	A	B	<input checked="" type="radio"/>	D