Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	239002 K1.03	
	Importance Rating	3.5	

Knowledge of the physical connections and/or cause- effect relationships between RELIEF/SAFETY VALVES and the following: Nuclear boiler instrument system

Proposed Question: RO Question # 1

When Reactor pressure instrumentation senses pressure has reached 1153 psig, which one of the following describes the expected response of both the Safety Relief Valves (SRVs) and the Safety Valves (SVs)?

A. Only one SRV will open.
NO SVs will open.

B. More than one but less than four SRVs will open.
NO SVs will open.

C. Four SRVs will open.
NO SVs will open.

D. Four SRVs will open.
Two SVs will open.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - With Reactor pressure at 1153 psig, this is above the set pressure of all four valves and all the valves will correspondingly open.

B. Incorrect - With Reactor pressure at 1153 psig, this is above the set pressure of all four valves and all the valves will correspondingly open.

C. Correct - The safety/relief valves are self actuating Target Rock Relief Valves set between 1095 and 1115 ± 11 psig per T.S. and have a capacity of 862,125 lbm/hr each at a reference pressure of 1080 psig. Each SRV is sized to relieve 10% of the main steam system flow. With Reactor pressure at 1153 psig, this is above the set pressure of all four valves and all the valves will correspondingly open.
The two self-actuating safety valves lift at 1240 ± 13 psig. The valves open when sufficient reactor pressure forces the valve upward against spring pressure. Spring pressure (and thus the lift setpoint) can be adjusted by a compression screw at the safety valve top. Because Reactor pressure has not reached 1240 psig the SVs will remain closed.

D. Incorrect - The two self-actuating safety valves lift at 1240 ± 13 psig. The valves open when sufficient reactor pressure forces the valve upward against spring pressure. Spring pressure (and thus the lift setpoint) can be adjusted by a compression screw at the safety valve top. Because Reactor pressure has not reached 1240 psig the SVs will remain closed.

Technical Reference(s): Tech Specs, 3.6.D.1 and Main Steam System Description pgs 12 and 13 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP, O-OR-02-04-01, EO-5 (As available)

Question Source: Bank # WTS Bank (River Bend)
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 3
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215003 K1.06	
	Importance Rating	3.9	

Knowledge of the physical connections and/or cause-effect relationships between INTERMEDIATE RANGE MONITOR (IRM) SYSTEM and the following: APRM SCRAM signals: Plant-Specific

Proposed Question: RO Question # 2

Given the following conditions:

- Control rod insertions are in progress for scheduled plant shutdown
- Reactor Mode Switch is in RUN
- Current reactor power is 9%.
- The "H" Intermediate Range Monitoring (IRM) Channel IRM function switch is OUT of OPERATE and has NOT been bypassed with the joystick
- All other IRMs indicate between 25 and 75 on the 0-125 scale

Which one of the following will cause a half scram?

A half scram will occur if …

A. APRM B fails downscale

B. APRM D fails downscale

C. IRM F fails upscale or inoperative

D. IRM G fails upscale or inoperative

Proposed Answer: A

Explanation (Optional):

A. Correct - A Scram signal is initiated on the associated RPS channel and rod block is inserted when any IRM module is unplugged, high voltage decreases below 95 percent or normal, or the IRM function switch is not in "OPERATE". However this scram is only BYPASSED with the Reactor Mode switch in "RUN" when the companion ARPM channel is not downscale. In this case with the H IRM inoperative because its function

switch is not in Operate a half scram will occur when B APRM goes downscale.

B. Incorrect - IRMs are still in normal range on R 10 and no half scram occurs. The companion APRM to IRM H is B. APRM D is the companion to IRM D.

C. Incorrect – With mode switch in Run IRM high/ inop scrams are bypassed except as explained in justification for B.

D. Incorrect - With mode switch in Run IRM high/ inop scrams are bypassed except as explained in justification for B.

Technical Reference(s): Procedure 2.2.65, Sect. 4.4, pg 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: LP O-RO-02-07-02, EO-10 (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 K2.02	
	Importance Rating	2.6	

Knowledge of electrical power supplies to the following: APRM channels

Proposed Question: RO Question # 3

Which one of the following describes the power supplies for the APRMs and LPRMs?

A. Buses Y-31 and Y-41supply their respective APRM channels
RPS Buses A and B supply their respective LPRM channels

B. RPS Buses A and B supply their respective APRM channels
Buses Y-31 and Y-41supply their respective LPRM channels

C. Bus Y-31 supplies APRM Channels A, C, and E and their respective LPRMs
Bus Y-41 supplies APRM Channels B, D, and F and their respective LPRMs

D. RPS Bus A supplies APRM Channels A, C, and E and their respective LPRMs
RPS Bus B supplies APRM Channels B, D, and F and their respective LPRMs

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Buses Y-31 and Y-41 supply power to systems required for plant safety except for the APRMs and LPRMs which are both supplied by the RPS Buses.

B. Incorrect – Buses Y-31 and Y-41 supply power to systems required for plant safety except for the APRMs and LPRMs which are both supplied by the RPS Buses.

C. Incorrect – Buses Y-31 and Y-41 supply power to systems required for plant safety except for the APRMs and LPRMs which are both supplied by the RPS Buses.

D. Correct - APRM Channels A, C, and E are powered from the same AC bus used for trip system A of the Reactor Protection System; APRM Channels B, D, and F are powered from the AC bus used for trip system B. The 120 volt AC bus used for a given APRM channel is the same as that used for the LPRMs providing inputs to that APRM.

Technical Reference(s): Procedure 2.2.67, Sect. 4.1.[2], pg 6 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-OR-02-07-04, 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 X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215004 K2.01	
	Importance Rating	2.6	

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

Proposed Question: RO Question # 4

Which one of the following would occur if 120 VAC panel Y-1 was lost during a reactor startup?

A. ONLY SRMs Detectors 'A' and 'C' can be moved from the C905 Panel

B. ONLY SRMs Detectors 'B' and 'D' can be moved from the C905 Panel

C. ALL of the SRMs Detectors can be selected from the C905 Panel but the detector drive motors CAN NOT be energized.

D. NONE of the SRMs Detectors can be selected from the C905 Panel but the detector drive motors CAN be energized

Proposed Answer: D

Explanation (Optional):

A. Incorrect. All SRM and IRM Detector drives cannot be selected

B. Incorrect. All SRM and IRM Detector drives cannot be selected

C. Incorrect. - All SRM and IRM Detector drives cannot be selected.

D. Correct. Y-1 supplies 120 VAC to SRM/IRM drive relay control

Technical Reference(s): 5.3.7, Page 6 and 11 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-01, EO# 8 (As available)

Question Source: Bank # TADs ID 327

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 K3.02	
	Importance Rating	3.8	

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

Proposed Question: RO Question # 5

The High Pressure Coolant Injection (HPCI) System is operating in the pressure control mode with the following:

- Reactor pressure is steady at 880 psig
- The HPCI Controller is in AUTO

Then the flow signal to the HPCI controller fails to zero.

Which one of the following describes how Reactor pressure and HPCI flow are affected by this failure?

	Reactor Pressure	Actual HPCI Flow
A.	Rises	Rises
B.	Rises	Lowers
C.	Lowers	Rises
D.	Lowers	Lowers

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Because reactor pressure would lower due to the increased steam demand.

B. Incorrect - Because reactor pressure would lower due to the increased steam demand and HPCI flow would rise as the controller attempted to increase flow.

C. Correct - In AUTO or BAL, flow demand signal, as determined by the FIC set point tape, controls turbine speed. The speed control circuit has a 0-4000 rpm range with the lowest flow corresponding to 2000 rpm. With the controller in AUTO (system operating to maintain the selected flowrate), a reduction in reactor pressure will result in HPCI's

control valve closing down to maintain the flow rate constant.
A loss of HPCI flow signal will result in the speed controller raising the output signal. This causes the HPCI turbine speed/flow to rise. HPCI will remain in operation at a higher actual flow than before the failure. The effect of the increased HPCI load will be a greater steam demand on the reactor lowering reactor pressure.

D. Incorrect - Because HPCI flow would rise due to the controller attempting to raise flow.

Technical Reference(s): Procedure 2.2.21.5, Att 2 and System Description Sect 7. Pg 24 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-03, EO-17 (As available)

Question Source: Bank # WTS (Cooper)
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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	205000 K3.01	
	Importance Rating	3.3	

Knowledge of the effect that a loss or malfunction of the SHUTDOWN COOLING SYSTEM (RHR SHUTDOWN COOLING MODE) will have on following: Reactor pressure

Proposed Question: RO Question # 6

Given the following:

- The plant is maneuvering to cold shutdown following an extended high power run
- Reactor pressure is 10 psig and steady
- The MSIVs are closed
- "A" RHR pump is in shutdown cooling in accordance with PNPS 2.2.19.1, RESIDUAL HEAT REMOVAL SYSTEM - SHUTDOWN COOLING MODE OF OPERATION

Which of the following malfunctions will result in a rise in reactor pressure? Assume no operator action.

Malfunction 1: MO-1001-16A, RHR HX A BYP VLV, fails full open.
Malfunction 2: Loss of all 125 VDC electrical power to the Group 3 Isolation Logic.
Malfunction 3: Loss of 120 VAC Safeguard bus Y-3.

A. Malfunction 1 only

B. Malfunction 3 only

C. Malfunctions 2 and 3 only

D. Malfunctions 1 and 2 only

Proposed Answer: A

Explanation (Optional):

A. Correct: MO-1001-16A, is the heat exchanger bypass valve. This valve failing open will cause a portion of the RHR flow to bypass the heat exchanger reducing the heat removed from the vessel, causing reactor pressure to rise.

B. Incorrect: PDC94-24, Shutdown Cooling Logic Improvements, changed the power supply to the PCIS logic for Group 3 SDC isolations from 120V AC (Y3/Y4) to 125V DC

(D4/D5). These relays were changed from "DE-ENERGIZE TO OPERATE" to "ENERGIZE TO OPERATE", which means on a loss of D4 and/or D5, the MO-1001-47 and MO-1001-50 SDC Isolation Valves will not close.

C. Incorrect: PDC94-24, Shutdown Cooling Logic Improvements, changed the power supply to the PCIS logic for Group 3 SDC isolations from 120V AC (Y3/Y4) to 125V DC (D4/D5). These relays were changed from "DE-ENERGIZE TO OPERATE" to "ENERGIZE TO OPERATE", which means on a loss of D4 and/or D5, the MO-1001-47 and MO-1001-50 SDC Isolation Valves will not close. Additionally, a loss of Y-3 will not cause SDC to isolate. Plausible in that it will cause the inboard isolation valves to close for Group 2 and Group 6 isolations and RBIS.

D. Incorrect: A loss of Y-3 will not cause SDC to isolate. Plausible in that it will cause the inboard isolation valves to close for Group 2 and Group 6 isolations and RBIS.

Technical Reference(s): PNPS 2.4.25, Loss of SDC, page 10. (Attach if not previously provided)
RHR Reference Text, pages 12 and 13 for a description of how the heat exchanger is operated to cooldown the reactor.

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 5
55.43

Comments:

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

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

Proposed Question: RO Question # 7

The plant is operating at 100% power when the "RCIC System Injection Mode Push Button" is depressed.

Which one of the following correctly describes the Reactor Core Isolation Cooling (RCIC) actions and the panel 904 indications?

A. The MANUAL INITIATION indicating lamp above the pushbutton will illuminate when the push button is depressed. The lamp will remain energized for 30 seconds and then extinguish. Provided that a Reactor Vessel low-low water level signal is present or occurs within the next 30 seconds, RCIC will start and inject.

B. The MANUAL INITIATION indicating lamp above the pushbutton will illuminate when the push button is depressed. The lamp will remain energized until the RCIC initiation is reset. RCIC will not automatically start until receipt of a Reactor Vessel low-low water level signal.

C. RCIC steam supply, injection, and other valves reposition, RCIC injection flow rises to 400 gpm. The MANUAL INITIATION indicating lamp above the pushbutton will illuminate when the push button is depressed. The lamp will remain energized until the RCIC initiation is reset pushbutton is depressed.

D. RCIC steam supply, injection, and other valves reposition, RCIC injection flow rises to 400 gpm. The MANUAL INITIATION indicating lamp above the pushbutton will illuminate when the push button is depressed. The lamp will remain energized for 30 seconds and then extinguish.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: The System will initiate as soon as the button is depressed. Pushing the button simulates a low-low water level signal.

B. Incorrect - RCIC will start RCIC in the full flow injection mode when the switch is depressed.

C. Incorrect - Once the switch is depressed, the control logic will seal in for 30 seconds. An indicator lamp will illuminate when the control logic is actuated. The lamp will remain energized during this start period (30 seconds). At the end of this period, the control logic will automatically reset and the lamp will de-energize.

D. Correct - IAW Procedure 2.2.25, The RCIC System control logic has been modified to add a single push button switch on Panel C904 which will start RCIC in the full flow injection mode when the switch is depressed. Once the switch is depressed, the control logic will seal in for 30 seconds. An indicator lamp will illuminate when the control logic is actuated. The lamp will remain energized during this start period (30 seconds). At the end of this period, the control logic will automatically reset and the lamp will de-energize. The system will be running at this time and will continue running until it is shut down by an Operator.

Technical Reference(s): Procedure 2.2.22.5, pages 13 and 14. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-04, EOs-7 & 9 (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	300000 K4.03	
	Importance Rating	2.8	

Knowledge of (INSTRUMENT AIR SYSTEM) design feature(s) and or interlocks which provide for the following: Securing of IAS upon loss of cooling water

Proposed Question: RO Question # 8

The plant is at 100% power with the following:

- A loss of Turbine Building Closed Cooling Water (TBCCW) occurs
- The Instrument Air Compressors (IACs) are operating in parallel control mode

Which of the following describes the IACs response with no operator action?

A. The K110 and K111 IACs trip on high discharge temperature, the K117 IAC starts and restores Instrument Air pressure.

B. The K110 and K111 IACs trip on low cooling water flow interlock, the K117 IAC starts and restores Instrument Air pressure.

C. The K111 and K117 IACs trip on low cooling water flow interlock, the K110 IAC trips on high discharge temperature.

D. The K110 and K111 IACs trip on high discharge temperature, the K117 IAC trips on high oil temperature, or high intercooler temperature.

Proposed Answer: A

Explanation (Optional):

A. Correct – With the IACs operating in parallel a trip of either lead compressor (110 or 111) will result in the other compressor (111 or 110) assuming the load. Both these compressors will trip on high discharge temperature with a loss of TBCCW. AS pressure lowers further IAC 117 the Diesel Air Compressor will start. IAC 117 does not use TBCCW for cooling water so it will auto start and restore IA pressure.

B. Incorrect - K110 and K111 IACs trip on high discharge temperature, these compressors do NOT have a trip on low TBCCW flow to the compressor.

C. Incorrect - IAC 117 the Diesel Air Compressor will start. IAC 117 does not use TBCCW for cooling water so it will continue to operate and maintain IA pressure. K111 trips on high discharge temperature, this compressor does NOT have a trip on low TBCCW flow to the compressor.

D. Incorrect - IAC 117 the Diesel Air Compressor will start. IAC 117 does not use TBCCW for cooling water so it will continue to operate and maintain IA pressure. Plausible in that the K-117 will trip on high oil temperature, or high intercooler temperature if either of these two conditions were to occur.

Technical Reference(s): Procedures 2.2.36, Sect 4.5, pgs 10-12. (Attach if not previously provided)
2.4.41, Sect. 2, pg 2

Proposed References to be provided to applicants during examination: None

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

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

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 4
55.43

Comments:

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

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

Proposed Question: RO Question # 9

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

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

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

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

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

Proposed Answer: C

Explanation (Optional):

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

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

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

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

Technical Reference(s): PNPS 5.3.8, Att 1, pg 9 (Attach if not previously provided)
System Description, pg 10, 16

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-06, EO-15.a & 19 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	263000 K3.02	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the D.C. ELECTRICAL DISTRIBUTION will have on following: Components using D.C. control power (i.e. breakers)

Proposed Question: RO Question # 10

Given the following:

- The plant is at rated conditions
- 125 VDC bus D-16 is lost

With these initial conditions:

- A manual scram is inserted
- The Main Turbine is tripped when generator load drops to less than 50 MWE
- No other operator actions are taken

Which one of the following lists all of the Feed and Condensate pumps that are still running ten seconds later?

A. "A" and "C" Reactor Feed pumps
"B" Condensate pump

B. "B" Reactor Feed pump
"A" and "C" Condensate pumps

C. No Reactor Feed Pumps
"B" Condensate pump

D. "A" and "C" Reactor Feed pumps
"A" and "C" Condensate pumps

Proposed Answer: B

Explanation (Optional):

A. Incorrect: These pumps will not be available because bus A1 will de-energize when the turbine trips due to the loss of associated breaker control power. Plausible in that this would be the response if D-17 were lost.

B. Correct: Normally when the turbine trips, A1, A3, and A5 transfer to the startup transformer. However the breakers associated with the transfer will be without control power. Therefore when the turbine trips, A1, A3, and A5 will be de-energized and associated loads will be lost. Bus A1 supplies "A" and "C" feed pumps and "B" condensate pump. The only remaining pumps will be "B" feed pump and "A" and "C" condensate pumps. The buses associated with these pumps are supplied with control power from the "B" battery (D-17).

C. Incorrect: "B" feed pump and "A" and "C" condensate pumps would be running. Plausible if the candidate thinks that D-16 supplies control power to bus A2. If so the candidate may also believe that when the other two condensate pumps are de-energized, that two reactor feed pumps will also trip on interlock. This is not true as the breakers associated with the condensate pumps do not trip as there is no control power. The bus supplying the pumps de-energize.

D. Incorrect: "A" and "C" feed pumps will not be running as they are powered from bus A1. Bus A1 de-energized when the turbine tripped.

Technical Reference(s): 5.3.11, LOSS OF ESSENTIAL DC BUS D16 OR D4 AND D36, page 11 (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

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Tier #	2
Group #	1
K/A #	261000 K6.08
Importance Rating	3.1

Knowledge of the effect that a loss or malfunction of the following will have on the STANDBY GAS TREATMENT SYSTEM: Reactor vessel level: Plant-Specific

Proposed Question: RO Question # 11

Given the following:

- The plant is at rated conditions.
- SBGT Fan 'A' Control Switch is in AUTO
- SBGT Fan 'B' Control Switch was inadvertently left in the MAINT position.

With these initial conditions, the reactor scrams and RPV level lowers to -10 inches before recovering. RPV level is now stable at +25 inches.

Which one of the following is correct regarding the response of the Standby Gas Treatment trains?

A. Train 'A' will start and remain running until manually shutdown. Train 'B' will not start.

B. Train 'B' will start and remain running until manually shutdown. Train 'A' will not start.

C. BOTH Train 'A' and Train 'B' will start and remain running until manually shutdown.

D. BOTH Train 'A' and Train 'B' will start. Train 'A' will automatically shutdown after 65 seconds.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Standby Gas Treatment will initiate when RPV level lowers to +12 inches. Both Trains will start. The function of the "MAINT" position of the 'B' Train Control switch is to prevent the 'B' Train from shutting down after 65 seconds as it normally does following a successful start of the 'A' Train.

B. Incorrect: The MAINT position does not prevent the 'A' Train from starting.

C. Correct: Standby Gas Treatment will initiate when RPV level lowers to +12 inches. Both Trains will start. The function of the "MAINT" position of the 'B' Train Control switch is to prevent the 'B' Train from shutting down after 65 seconds.

D. Incorrect: The 'A' Train will not automatically shutdown.

Technical Reference(s): PNPS 2.2.50, page 8 and page 10. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-03, EO-4 & 10 (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 K6.08	
	Importance Rating	3.5	

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

Proposed Question: RO Question # 12

During a plant startup the following conditions exist:

- The Reactor Mode Switch is in Startup
- MSIVs are open
- MSIV Inboard Drain Isolation Valve MO-220-1 is open
- MSIV Outboard Drains Isolation Valve MO-220-2 is open

Which one of the following identifies the Primary Containment Isolation System response to opening an EPA breaker on the output of the A RPS MG set?

A. There is no effect on the Group 1 isolation logic; both MO-220-1 and MO-220-2 remain open.

B. The AC solenoids on the Inboard MSIVs and the DC solenoids on the Outboard MSIVs de-energize. MO-220-1 closes.

C. The DC solenoids on the Inboard MSIVs and the AC solenoids on the Outboard MSIVs de-energize. MO-220-2 closes.

D. One half the Group 1 isolation logic is de-energized, however no solenoids are de-energized and no isolations occur.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – One half the logic is de-energized

B. Incorrect - No solenoids de-energize and no valve motion occurs

C. Incorrect - No solenoids de-energize and no valve motion occurs

D. Correct – The loss of RPS A will de-energize the "A" and "C" logic inputs to the Group 1 isolation logic this will cause a loss of half the relays, however each solenoids logic (AC and DC) require one out of two taken twice to de-energize a solenoid, therefore although half the logic is de-energized no actions occur and no valve motion occur.

Technical Reference(s): PNP 2.2.79, Sect 7.1.4, pg 17
PCIS SD, pages 11 and 12 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-10, EO-11.a (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 A1.08	
	Importance Rating	3.7	

Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) controls including: Emergency generator loading

Proposed Question: RO Question # 13

Given the following conditions:

- A LOCA is in progress
- Drywell pressure is 18 psig
- RHR Pumps A and C are injecting into the RPV
- RHR Pump B is in containment spray mode
- RHR Loop Cross-Tie Valve, MO-1001-19, has been closed
- RHR Pump D has been manually shutdown and its white override light is illuminated.

Two minutes later, a loss of offsite power occurs, de-energizing both the Startup and the Shutdown Transformers.

- "A" EDG output breaker fails to close
- "B" EDG output breaker closes and the diesel loads as designed

Which one of the following identifies the RHR pump status after power is restored from the EDG?

A. RHR Pump B is running in the injection mode. RHR pump D is not running.

B. RHR Pumps B and D are running in the containment spray mode only.

C. RHR Pump B is running in both the injection and containment spray modes. RHR pump D is not running.

D. RHR Pumps B and D are running in both the injection and containment spray modes

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The B EDG re-energizes A6 after a time delay. The "B" RHR pump will start

5 seconds after the bus is re-energized. The "D" RHR pump will start 10 seconds after the bus is re-energized. Although the pump was manually shutdown the bus power monitoring circuit will defeat the operator's manual over ride and the pump restarts.

B. Correct - The white light will illuminate and remain lit if the operator manually shuts down an RHR pump with an auto start signal present (LPCI initiation signal). When offsite power is lost the operating RHR pumps will trip. The B EDG re-energizes A6 after a time delay. The "B" RHR pump will start 5 seconds after the bus is re-energized. The "D" RHR pump will start 10 seconds after the bus is re-energized. Although the pump was manually shutdown the bus power monitoring circuit will defeat the operator's manual over ride and the pump restarts. Loop select initially selected loop A for injection and was sealed in. It will not re-initiate because the logic never lost power (DC power). Because the containment spray valves remain open RHR Pumps B and D will operate in Containment Spray. There are no auto opening signals for the Cross-Tie valve.

C. Incorrect - The B EDG re-energizes A6 after a time delay. The "B" RHR pump will start 5 seconds after the bus is re-energized. The "D" RHR pump will start 10 seconds after the bus is re-energized.

D. Incorrect - The LPCI loop selection logic is DC powered and therefore it never lost power. Since it selected the "A" loop initially and seals in, loop "A" is still selected for injection and the "B" loop injection valves do not open. With the Cross-Tie closed and the containment spray valves open the B and D Pumps will operate in Containment Spray only (There are no auto opening signals for the Cross-Tie valve.)

Technical Reference(s): RHR System Description, pg 26 and 27 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-01, EO-9 (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	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A1.04	
	Importance Rating	3.7	

Ability to predict and/or monitor changes in parameters associated with operating the LOW PRESSURE CORE SPRAY SYSTEM controls including: Reactor pressure

Proposed Question: RO Question # 14

A LOCA occurred resulting in the following:

- RPV pressure is 200 psig
- Drywell pressure is 3.5 psig
- RPV level is -40 inches and rising
- HPCI is injecting into the RPV
- The operator attempts to close Core Spray Injection Valves MO-1400-24A, 24B, 25A, and 25B

How will the Core Spray Injection Valves respond?

	Injection Valves 24A & B will:	**Injection Valves 25A & B will:**
A.	remain open	remain open
B.	remain open	close
C.	close	remain open
D.	close	close

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The answers are combinations of the open and closed positions that are incorrect.

B. Correct - LPCS will initiate and the pump will start when RPV level is <-46 inches in conjunction with RPV pressure being less than 400, or drywell pressure is > 2.2.psig. The LPCs Injection Valve will open when reactor pressure drops below the 400 psig interlock provided an initiation signal is present. The Core Spray Loop A and B Injection Valves, MO-1400-25A and MO-1400-25B, can be manually closed due to Operator action with an initiation signal present. Both 24A & B valves receive an open signal regardless of position if a system initiation signal is received, and cannot be shut until the initiating signal is cleared.

C. Incorrect: The answers are combinations of the open and closed positions that are incorrect.

D. Incorrect: The answers are combinations of the open and closed positions that are incorrect.

Technical Reference(s): PNP 2.2.20, Sect 4.3, pg 9 and Sect 7.2, pg 15 System Description pg 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-02, EO-4 (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K6.01	
	Importance Rating	3.9	

Knowledge of the effect that a loss or malfunction of the following will have on the AUTOMATIC DEPRESSURIZATION SYSTEM : RHR/LPCI system pressure

Proposed Question: RO Question # 15

Given the following:

- A LOCA has occurred
- ADS has automatically initiated and all SRVs are open
- Reactor pressure is 300 psig and lowering
- All low pressure ECCS pumps are starting to inject

Then .....

- A complete loss of off-site power occurs
- All 4160 VAC buses de-energize
- Four seconds later, both EDG output breakers close and A-5 and A-6 re-energize.

Based on the sequence above, the four ADS valves _____

A. Closed and then reopened as soon as an ECCS pump re-started

B. Closed and then reopened as soon as A-5 and A-6 re-energized

C. Closed and remained closed

D. Remained open

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Valves remain open. Plausible in that "a pump running signal" based on ECCS pump discharge pressure is required to initiate the logic. The discharge pressure signal was lost once the buses de-energized following the loss of off-site power.

B. Incorrect: valves remain open. Plausible in that there are bus monitoring circuits in the logic but these circuits are monitoring DC bus status.

C. Incorrect: Valves remain open. Plausible if the candidate believes that ADS has already performed its function since the LP ECCS will inject and is no longer required.

D. Correct: Per PNPS 2.2.23, once the ADS RVs are opened, depressurization will continue even if the RHR/Core Spray pump running signal is lost.

Technical Reference(s): PNPS 2.2.23, ADS, page 10, item 4.2 [1] (d) (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	400000 A2.03	
	Importance Rating	2.9	

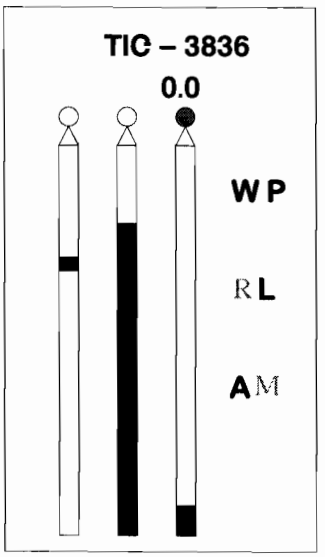
Ability to (a) predict the impacts of the following on the CCWS and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation: High/low CCW temperature

Proposed Question: RO Question # 16

Given the following:

- The plant is at rated conditions
- 'A' Fuel Pool Cooling Heat Exchanger was placed in service 5 minutes ago
- The BOP operator reports that RBCCW Loop 'A' temperature has risen from 70 degrees to 82 degrees.
- 'A' and "B' RBCCW pumps are in service.
- Indications on TIC – 3836, RBCCW Loop 'A' Temperature Controller are as shown in the drawing below.

Which one of the following is correct regarding the controller operation and what actions are required to restore RBCCW temperature to normal?



A. The controller has NOT responded correctly to the increase in heat load. Place the controller in manual and increase the controller output.

B. The controller has NOT responded correctly to the increase in heat load. Manually close MO-4084, Loop 'A' RBCCW Heat Exchanger Bypass Valve, to increase RBCCW flow through the heat exchanger.

C. The controller IS responding correctly to the increase in heat load. Jog open MO-3800 'A' RBCCW Heat Exchanger SSW Outlet Valve to increase SSW flow.

D. The controller IS responding correctly to the increase in heat load. Start the third RBCCW pump and increase RBCCW flow through the heat exchanger to 5000 gpm.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The controller is responding as expected to temperature being higher than the setpoint (70 degrees). The controller controls the position of a bypass valve. As temperature comes up the controller lowers the output which will close down on the bypass valve forcing more RBCCW flow thru the heat exchanger. Plausible in that the output of the controller is zero.

B. Incorrect: The controller is responding as expected to temperature being higher than the setpoint (70 degrees). The controller controls the position of a bypass valve. As temperature comes up the controller lowers the output which will close down on the bypass valve forcing more RBCCW flow thru the heat exchanger. Plausible in that the output of the controller is zero.

C. Correct: As discussed above the controller is responding normally. Additional SSW flow is required to lower the RBCCW temperature. Per PNPS 2.2.32 SSW, MO-3800 'A' RBCCW Heat Exchanger SSW Outlet Valve is adjusted as required based on plant conditions to control temperature.

D. Incorrect: Per PNPS 2.2.32 SSW, MO-3800 'A' RBCCW Heat Exchanger SSW Outlet Valve is adjusted as required based on plant conditions to control temperature. Additionally, PNPS 2.2.30, page 14, limits flow through the heat exchanger to 4000 gpm.

Technical Reference(s): PNPS 2.2.32, SSW System, page 23.
RBCCW Reference Text, pages 10 and 32 for a description of controller operation.
(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		
	Comprehension or Analysis		X

10 CFR Part 55 Content:	55.41	4
	55.43	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262001 A3.03	
	Importance Rating	3.4	

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

Proposed Question: RO Question # 17

The plant was operating at 100% power with the "B" CRD pump in service. Subsequently, a valid LOCA signal generated a scram. The plant responded as expected EXCEPT the startup transformer feeder breaker to bus A-5 failed to close. The A-5 bus has been automatically energized from the shutdown transformer as designed.

Which ONE of the following describes the status/availability of the CRD pumps?

The B CRD pump is ...

A. running, the A CRD pump cannot be started due to load shed signal.

B. running, the A CRD pump can be started since no load shed signal was generated.

C. NOT running, the A and B CRD pumps CANNOT be started due to a load shed signal.

D. NOT running, the A and B CRD pumps can be started since no load shed signal was generated.

Proposed Answer: A

Explanation (Optional):

A. Correct – The A CRD pump is powered from A-5, when the startup transformer failed to pick up the bus and a LOCA signal was generated a load shed occurred on the bus. The A CRD pump cannot be restarted until the load shed signals are cleared. The B CRD pump is powered from bus A-6 and since this bus was powered from the startup transformer a load shed did not occur, consequently the B CRD pump is still running.

B. Incorrect - The A CRD pump is powered from A-5, when the startup transformer failed to pick up the bus and a LOCA signal was generated a load shed occurred on the bus. The A CRD pump cannot be restarted until the load shed signals are cleared.

C. Incorrect - The B CRD pump is powered from bus A-6 and since this bus was powered from the startup transformer a load shed did not occur, consequently the B CRD pump is still running.

D. Incorrect - The B CRD pump is powered from bus A-6 and since this bus was powered from the startup transformer a load shed did not occur, consequently the B CRD pump is still running. The A CRD pump is powered from A-5, when the startup transformer failed to pick up the bus and a LOCA signal was generated a load shed occurred on the bus. The A CRD pump cannot be restarted until the load shed signals are cleared.

Technical Reference(s): 2.4.16, Att. 8, pgs 25 & 26 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-11, EO-2c
O-RO-02-09-08, EO-6 (As available)

Question Source: Bank # TADs ID: 3310
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam:

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

10 CFR Part 55 Content: 55.41 6
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	264000 A3.03	
	Importance Rating	3.4	

Ability to monitor automatic operations of the EMERGENCY GENERATORS (DIESEL/JET) including: Indicating lights, meters, and recorders

Proposed Question: RO Question # 18

A diesel generator (D/G) is supplying an electrical bus in parallel with the grid.

Assuming D/G terminal voltage does not change, how are the D/G KVAR and D/G Amps affected if the D/G governor is placed in the RAISE position for five (5) seconds?

	D/G KVAR	D/G Amps
A.	No change	Rise
B.	No change	No change
C.	Rise	Rise
D.	Rise	No change

Proposed Answer: A

Explanation (Optional):

A. Correct – The D/G is loaded using the Governor Speed Control, increasing the speed setpoint will have a directly impact real power (kW) raising the output amps of the D/G. Reactive loading (KVAR) is controlled using the Voltage Regulator Setpoint Adjuster which is not adjusted in this question.

B. Incorrect - The D/G is loaded using the Governor Speed Control, increasing the speed setpoint will have a directly impact real power (kW) raising the output amps of the D/G

C. Incorrect - Reactive loading (KVAR) is controlled using the Voltage Regulator Setpoint Adjuster which is not adjusted in this question.

D. Incorrect - The D/G is loaded using the Governor Speed Control, increasing the speed setpoint will have a directly impact real power (kW) raising the output amps of the D/G. Reactive loading (KVAR) is controlled using the Voltage Regulator Setpoint Adjuster

which is not adjusted in this question.

Technical Reference(s): 2.2.8, Sect 7.5.1, pg 40 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-01-04-05, EO-19 (As available)

Question Source: Bank # TADs ID: 5126
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

Comments:

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

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

Proposed Question: RO Question # 19

Given the following:

- The plant is at 30% power with a normal electrical distribution lineup;
- 480 VAC Load Center B-6 is lost when the supply breaker on B-1, B52-102, trips due to a breaker fault;
- Operators report the following indications regarding 120 VAC Vital Bus Y-2:
    - Alarm Y-2 AUTOMATIC TRANSFER, C3RC-A2, has annunciated
    - The Y-2 potential indicating light on C3 extinguished briefly but is now lit.

Operators have stabilized the plant and are now making preparations to re-energize B6 from 480 VAC Load Center B-2.

Based on the above information which one of the following is correct regarding

(1) The initial response of Y-2 to the transient

AND

(2) Any automatic response of Y-2 when B-6 is re-energized?

A. (1) Y-2 responded as designed
(2) Y-2 will remain on its alternate power supply, B-15

B. (1) Y-2 did NOT respond as designed
(2) Y-2 will remain on its alternate power supply, B-15

C. (1) Y-2 responded as designed
(2) Y-2 will automatically transfer back to its preferred power supply, the Y-2 MG Set

D. (1) Y-2 did NOT respond as designed
(2) Y-2 will automatically transfer back to its preferred power supply, the Y-2 MG Set

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Y-2 is normally powered from a MG set that has both a DC and AC driver. The AC driver, powered from B-6 is normally powering the MG set. If B-6 power is lost, the DC motor will then power the MG set. A large flywheel on the MG maintains Y-2 voltage and frequency as the drivers shift from AC to DC. If the MG set fails or DC power is not available and Y-2 de-energizes, Y-2 will transfer to its alternate source, B-15. In this case, Y-2 should not have de-energized and auto transferred indicating a failure of the DC motor to maintain the MG set energized.

B. Correct: Y-2 should not have transferred and should have remained energized during the transient via the DC supply to the MG set. Y-2 will not automatically transfer back to its preferred source when B-6 is re-energized. Any Y-2 automatic transfer to B-15 must be manually reset.

C. Incorrect: Y-2 should not have transferred and must be manually re-aligned to its preferred source following the automatic transfer.

D. Incorrect: Y-2 must be manually re-aligned to its preferred source following an automatic transfer. Plausible in that if the MG set is being powering by the DC motor following a loss of B-6 power, and B-6 is subsequently restored, the MG set will automatically shift back to AC power. Additionally Y-1 will also auto transfer back to its preferred source if an automatic transfer to the alternate has occurred.

Technical Reference(s): PNPS 2.2.16, 120/240V AC VITAL SERVICES INSTRUMENT POWER SUPPLY (Y2), page 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: 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		

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 A4.04	
	Importance Rating	3.7	

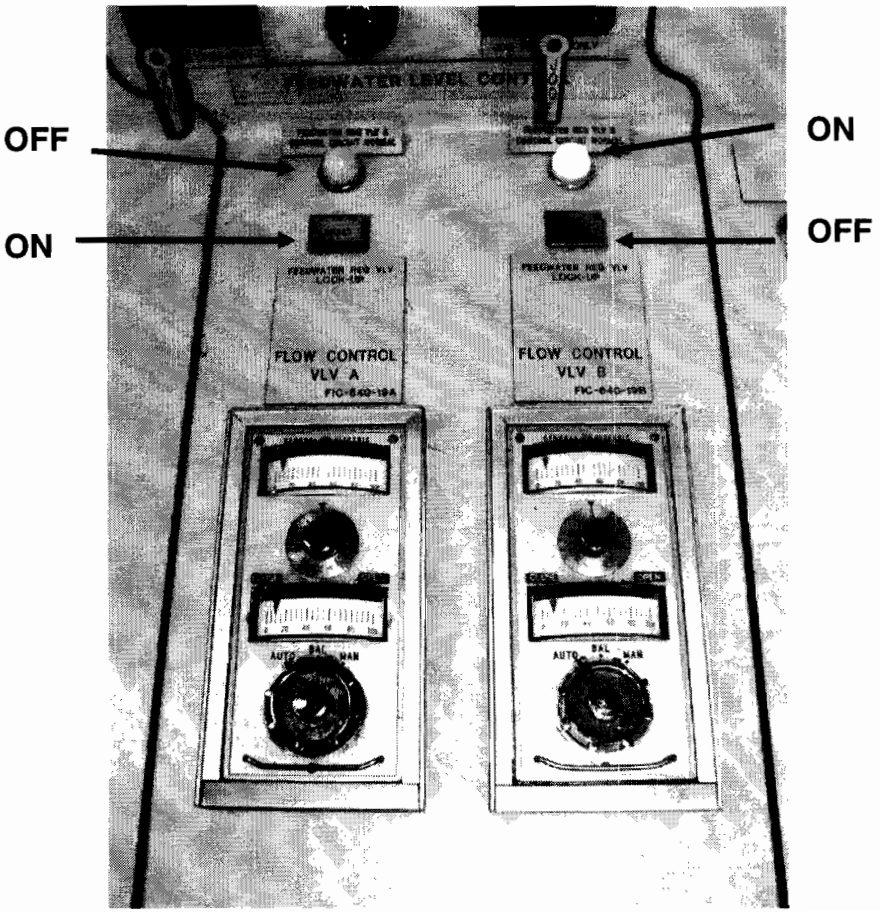
Ability to manually operate and/or monitor in the control room: FWRV lockup reset controls

Proposed Question: RO Question # 20

The plant is at rated conditions. Feedwater Level Control is in Master Auto and set to control at +30 inches. Then, operators note the following:

- "A" Feed Line Flow (FI-640-24A) is 6 Mlbm/hr and slowly rising
- "B" Feed Line Flow (FI-640-24B) is 3 Mlbm/hr and slowly lowering
- Reactor water level remains at +30 inches.

Other feed water level control indications are as shown in the picture. Which one of the following is consistent with these indications?



A. The “A” M/A Station has failed

B. The “B” M/A Station has failed

C. The “A” feed reg valve is locked up and is drifting open

D. The “B” feed reg valve is locked up and is drifting closed

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The M/A stations are operating however the “A” Feed Regulating Valve has locked up and slowly drifting open

B. Incorrect – The M/A stations are operating however the “A “Feed Regulating Valve has locked up and slowly drifting open

C. Correct - On a loss of air to the Feedwater Regulating Valve(s), the red FEED REG VLV LOCK-UP RESET light(s) will illuminate and the white FEED REG VLV A (B) CONTROL CIRCUIT NORMAL light(s) will extinguish. The A Feed Regulating Valve has failed and lock up and is drifting open causing A feed line flow to rise, level will temporarily remain under control as the B feedwater flow is lowered.

D. Incorrect - The A Feed Regulating Valve has failed and lock up and is drifting open causing A feed line flow to rise, level will temporarily remain under control as the B feedwater flow is lowered.

Technical Reference(s): PNPS 2.4.49, pg 20, sect. 5.0, [6] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-04-10, EO-18 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam: No

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

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 2.4.31	
	Importance Rating	4.2	

Emergency Procedures / Plan: Knowledge of annunciator alarms, indications, or response procedures (RPS)

Proposed Question: RO Question # 21

Given the following:

- PNPS is at rated conditions when an inadvertent MSIV isolation results in RPV pressure rising rapidly.
- All control rods automatically insert.
- While stabilizing the plant following the trip, the 905 panel operator observes the following:

  - All "A" RPS Group solenoid lights: ON
  - All "B" RPS Group solenoid lights: OFF
  - Alarm ATWS DIVISION ONE TRIPPED (C905L-A5): IN ALARM
  - Alarm ATWS DIVISION TWO TRIPPED (C905L-E5) CLEAR
  - "A" Recirc MG set: TRIPPED
  - "B" Recirc MG set: 26% SPEED

Based on the above

(1) How many ARI valves actuated to cause control rod insertion

AND

(2) Did the ATWS DIVISION ONE circuitry respond as designed and If not, why not?

A. (1) Two
(2) ATWS Division One circuitry responded as designed.

B. (1) Two
(2) ATWS Division One circuitry failed to trip the "B" recirc MG set

C. (1) One
(2) ATWS Division One circuitry responded as designed.

D. (1) One
(2) ATWS Division One circuitry failed to trip the "B" recirc MG set

Proposed Answer: D

Explanation (Optional):

A. Incorrect: One ARI valve is associated with each division. Only one ARI valve is required to depressurize the scram air header. Since only Division One tripped, only one ARI valve repositioned. Additionally, either division will trip both recirc pumps. The "B" MG set should have been tripped by the Division one logic.

B. Incorrect: Only one ARI valve repositioned.

C. Incorrect: The "B" MG set should have been tripped by the Division one logic.

D. Correct: Each division will energize its respective ARI valve which is sufficient to de-energize the header. Each division will also trip BOTH recirc pumps. The "B" Recirc pump should have tripped.

Technical Reference(s): ATWS System Reference Text, page 14 (Attach if not previously provided)
ARP C905L-A5

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # LOR Bank #359
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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	206000 2.2.40	
	Importance Rating	3.4	

Equipment Control: Ability to apply technical specifications for a system (HPCI).

Proposed Question: RO Question # 22

Given the following:

- The plant is operating at full power.
- The Condensate Storage Tank (CST) low level switches have just been declared inoperable.

Which one of the following is correct regarding the impact on HPCI and RCIC?

A. HPCI and RCIC remain operable, due to a redundant suction source.

B. Only HPCI is inoperable, due to HPCI suction valve interlock being inoperable.

C. Only RCIC is inoperable, due to RCIC suction valve interlock being inoperable.

D. HPCI and RCIC are inoperable, due to HPCI and RCIC suction valve interlocks being inoperable.

Proposed Answer: B

Explanation (Optional):

A. Incorrect - Both HPCI and RCIC have a redundant suction source using the Torus; however HPCI has an automatic suction transfer when water in the CST falls below a predetermined level. This suction interlock requires these CST low level switches be operable. RCIC does not have this automatic suction swap and therefore is unaffected by these switches becoming inoperable.

B. Correct - Both HPCI and RCIC have a redundant suction source using the Torus, however HPCI has an automatic suction transfer when water in the CST falls below a predetermined level. This suction interlock requires these CST low level switches be operable.

C. Incorrect - RCIC does not have this automatic suction swap and therefore is unaffected by these switches becoming inoperable.

D. Incorrect - RCIC does not have this automatic suction swap and therefore is unaffected by these switches becoming inoperable.

Technical Reference(s): T.S. 3.2.B, Table 3.2.B, pg 3/4.2-16
PNPS 2.2.21, Sect 4.3, pg 8 and Sect. 5.2.[4], pg 16 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-09-03, EO-26 (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	212000 A1.08	
	Importance Rating	3.4	

Ability to predict and/or monitor changes in parameters associated with operating the REACTOR PROTECTION SYSTEM controls including: Valve position

Proposed Question: RO Question # 23

A turbine trip from full power has caused a reactor scram. RPV level lowered to -20 inches during the initial transient but has been restored to the normal operating band. The scram has NOT BEEN RESET.

Which one of the following correctly describes the status of the RPS Backup Scram valves in this plant condition?

Both Backup Scram valves should be ...

A. energized and aligned to vent the air header

B. de-energized and aligned to vent the air header

C. energized and NOT aligned to vent the air header

D. de-energized and NOT aligned to vent the air header

Proposed Answer: A

Explanation (Optional):

A. Correct - Whenever a Reactor Scram occurs, both Backup Scram Valve solenoids (normally de-energized) are energized and instrument air is blocked and vented at this point.

B. Incorrect - Whenever a Reactor Scram occurs, both Backup Scram Valve solenoids (normally de-energized) are energized.

C. Incorrect - Whenever a Reactor Scram occurs, both Backup Scram Valve solenoids (normally de-energized) are energized and instrument air is blocked and vented at this point.

D. Incorrect - Whenever a Reactor Scram occurs, both Backup Scram Valve solenoids (normally de-energized) are energized and instrument air is blocked and vented at this point.

Technical Reference(s): PNPS 2.2.79, Sect. 4.2 [2], pg 9 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-07-07, EO-3.f (As available)

Question Source: Bank # TADs ID: 12604

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	209001 A3.02	
	Importance Rating	3.8	

Ability to monitor automatic operations of the LOW PRESSURE CORE SPRAY SYSTEM including: Pump start

Proposed Question: RO Question # 24

A feedwater line break outside containment results in a lowering RPV level. Reactor Pressure is currently 800 psig and stable.

Which one of the following correctly describes the automatic response of the Core Spray System?

The Core Spray pumps will start…

A. immediately after RPV level lowers to -46 inches.

B. if RPV level lowers and remains less than -46 inches for a minimum of 11 minutes.

C. if RPV level lowers and remains less than -46 inches for a minimum of 13 minutes.

D. once RPV level lowers to -46 inches AND RPV pressure is less than 400 psig AND 11 minutes have elapsed.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Given the current reactor pressure the Core Spray pumps will not start without a high drywell pressure signal or until the high Drywell pressure bypass timer times out.

B. Correct - The Core Spray System will start automatically in response to any of three signals:
(1) +2.22 psig Drywell pressure (valves will not open until Reactor pressure is less than 395 to 405 psig).
(2) -46.3 inches RPV water level and with RPV pressure below 395 to 405 psig.
(3) -46.3 inches RPV water level and expiration of the high Drywell pressure bypass timer (+9 to +15.4 minutes) (nominal setting is considered 11 minutes).

C. Incorrect: The pumps will start at the 11 minute point. Plausible in that this is the time delay associated with an ADS blowdown when utilizing the drywell high pressure

bypass feature.

D. Incorrect – The 11 minute timer is not required with the other conditions present.

Technical Reference(s): PNPS 2.2.20, Sect. 4.2 [1], pg 8 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

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

Question Source: Bank # TADs ID: 720
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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	218000 K3.01	
	Importance Rating	4.4	

Knowledge of the effect that a loss or malfunction of the AUTOMATIC DEPRESSURIZATION SYSTEM will have on following: Restoration of reactor water level after a break that does not depressurize the reactor when required

Proposed Question: RO Question # 25

PNPS is at rated power when a loss of 125 VDC panel D-4 occurs. One of the many alarms that annunciate is the following ADS alarm:

ADS POWER FAILURE (C903L-A1)

While in this condition a small break LOCA inside the drywell results in drywell pressure rising to 3.4 psig and lowering RPV water level.

Given that the MSIVs have just closed on low vessel level and that no high pressure injection sources are available, which one of the following correctly states when RPV level will be recovered?

Low pressure ECCS will recover level when ADS ...

A. Initiates two minutes later. All four SRVs will open.

B. Initiates two minutes later. ONLY the "A" and "C" SRVs will open.

C. Initiates eleven minutes later. All four SRVs will open.

D. Initiates eleven minutes later. ONLY the "A" and "C" SRVs will open.

Proposed Answer: A

Explanation (Optional):

A. Correct - A 105 second ADS timer is actuated by either a simultaneous occurrence of high drywell pressure (2.2 psig) and low-low RPV water level (-46 in.). Since the MSIVs also close at -46 inches, conditions have been met to start the timer. C903L-A1 alarms when a loss of 125 VDC panel D-4 occurs. This power loss disables ADS Logic Train "A"; however ADS Logic Train "B" is operable. The loss of any battery affects only one two minute timing circuit. Each relief valve is powered by DC from either station battery through auto-transfer switches. Therefore the ADS valves will open when the two

minute timer times out.

B. Incorrect – All four valves will open because each relief valve is powered by DC from either station battery through auto-transfer switches.

C. Incorrect - The loss of any battery affects only one two minute timing circuit. Each relief valve is powered by DC from either station battery through auto-transfer switches. Therefore the ADS valves will open when the two minute timer times out.

D. Incorrect - The loss of any battery affects only one two minute timing circuit. Each relief valve is powered by DC from either station battery through auto-transfer switches. Therefore the ADS valves will open when the two minute timer times out.

Technical Reference(s): ARP-903L-A-1
5.3.11, Sect. 2.0, [5](i) pg 3 (Attach if not previously provided)
ADS System Description

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RE-02-09-05, EO-26 (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

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 # 26

Given the following:

- Following a total loss of all off-site power RCIC is being used to control RPV water level.
- RCIC is operating in Automatic at rated flow
- Then, an over-voltage transient on "A" 125 VDC system results in alarm RCIC INVERTER FAILURE, C904L A4.
- Three (3) seconds later, voltage on the "A" 125 VDC system returns to normal.

(1) How does the INVERTER FAILURE impact the operation of the RCIC Flow Controller

AND

(2) How will the inverter respond when the voltage on the "A" 125 VDC system returns to normal?

A. (1) The output of the controller will fail to MAXIMUM demand resulting in an increase in RCIC speed
(2) The inverter will auto reset and the controller output will return to normal.

B. (1) The output of the controller will fail to MINIMUM demand resulting in a decrease in RCIC speed
(2) The inverter will auto reset and the controller output will return to normal.

C. (1) The output of the controller will fail to MAXIMUM demand resulting in an increase in RCIC speed
(2) The inverter will remain tripped until manually reset at the C904 panel.

D. (1) The output of the controller will fail to MINIMUM demand resulting in a decrease in RCIC speed
(2) The inverter will remain tripped until manually reset at the C904 panel.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The controller loses power and the output will fail to minimum, resulting in a reduction in turbine speed and flow.

B. Correct: The control circuitry for the RCIC System is 115V AC supplied by an inverter from the 125V DC Bus "A". A high voltage input condition (approximately 160V DC) will trip the inverter. The unit will automatically reset after the input voltage conditions return to normal with an approximately 3-second time delay. The loss of the 115V AC to the control circuitry will cause a reduction in the flow demand signal to the turbine.

C. Incorrect: The Inverter will auto reset. Plausible in that an earlier version of the inverter required a manual rest.

D. Incorrect: The Inverter will auto reset. Plausible in that an earlier version of the inverter required a manual rest.

Technical Reference(s): REACTOR CORE ISOLATION
COOLING SYSTEM (RCIC)
System Reference Text, page 11. (Attach if not previously provided)
ARP for C904L-A4

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 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	215002 K1.03	
	Importance Rating	3.2	

Knowledge of the physical connections and/or cause- effect relationships between ROD BLOCK MONITOR SYSTEM and the following: Reactor manual control: BWR-3,4,5

Proposed Question: RO Question # 27

Given the following:

- Reactor power is 40%
- A central control rod is selected for withdraw

Which one of the following will result in the Reactor Manual Control System imposing a rod withdraw block?

A. RBM A is indicating a value of 112%.

B. Five LPRMs currently feeding RBM B are bypassed.

C. RBM B switch S-2 is placed in the "COUNT" position.

D. Reference APRM to RBM A has drifted downward to 20%.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: At 40% power the RBM High trip setpoint is 120%. Plausible in that at a higher power, a value of 112% would cause a block (different setpoints based on reactor power).

B. Correct: A RBM INOP trip is generated when 50% of the LPRMs feeding the RBM are bypassed. Since this is a central control rod, there are 4 LPRM strings available and therefore 8 LPRMs are assigned.

C. Incorrect: The switch S-2 can be positioned without causing a rod block. Plausible in that switch S-1 cannot

D. Incorrect: The APRM drifting downward will result in an automatic bypass of the RBM, not an INOP trip

Technical Reference(s): RBM Reference Text, Figure 2 and pages 7 and 8 for LPRM assignments. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # TADS # 229

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 2

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	202002 A3.03	
	Importance Rating	3.1	

Ability to monitor automatic operations of the RECIRCULATION FLOW CONTROL SYSTEM including: Scoop tube operation: BWR-2,3,4

K/A Justification: The recirc flow controller controls the position of the scoop tube which in turn controls MG set speed. The #2 speed limiter initiates a runback by immediately changing the controller's output (demand signal to the scoop tube) to 44%. The operator monitors and verifies the response of the scoop tube via the Middle Bar Chart by observing changes in MG set speed.

Proposed Question: RO Question # 28

The "B" recirc MG set speed controller on the 904 panel is in MANUAL mode with the following initial indications on the controller:

- Left Bar Chart (Operator Setpoint): 70%
- Middle Bar Chart (MG Set Speed): 70%
- Right Bar Chart (Output): 70%

Given these initial conditions, which one of the following correctly describes the indications on the controller following a runback to the #2 speed limiter?

A. Left Bar Chart: 44%
Middle Bar Chart: 44%
Right Bar Chart: 44%

B. Left Bar Chart: 44%
Middle Bar Chart: 44%
Right Bar Chart: 70%

C. Left Bar Chart: 70%
Middle Bar Chart: 44%
Right Bar Chart: 44%

D. Left Bar Chart: 70%
Middle Bar Chart: 44%
Right Bar Chart: 70%

Proposed Answer: C

Explanation (Optional):

A. Incorrect – The Left Bar is the Operator Setpoint and with the controller in Manual, will not automatically respond.

B. Incorrect – The Left Bar is the Operator Setpoint and with the controller in Manual, will not automatically respond. Additionally, the right bar chart is the output of the controller. When the limiter is activated, the output of the controller will immediately drop to 44%. This will in-turn reduce the MG set speed.

C. Correct – No. 2 function will override controller output and will limit the speed demand signal to 44%, not subject to rate limiting. The left bar, the operator setpoint will not be changed by the runback to 44#. The center bar graph for controller indicates actual speed indication which would indicate 44% based on the #2 Speed Limiter. The right bar, indicating the controller output will indicate the new controller demand to the scoop tube, which would be 44% for the #2 Speed Limiter.

D. Incorrect – The right bar chart is the output of the controller. When the limiter is activated, the output of the controller will immediately drop to 44%. This will in-turn reduce the MG set speed.

Technical Reference(s): PNPS 2.2.84, Sect. 4.2.4, pgs 16 & 17, Att. 7, pg 107 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-06-02, EO-16b (As available)

Question Source: Bank # TADs ID: 3236

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	259001 K3.06	
	Importance Rating	3.1	

Knowledge of the effect that a loss or malfunction of the REACTOR FEEDWATER SYSTEM will have on following: Core inlet subcooling

Proposed Question: RO Question # 29

The plant is operating at 100% power when a partial loss of feedwater heating occurs. Feedwater temperature lowers by 15 degrees and stabilizes.

Which one of the following is the effect on core inlet subcooling and what Immediate Action is required by PNPS 2.4.150, Loss of Feedwater Heating?

	Core inlet subcooling	Lower Reactor power to:
A.	increases	<25%
B.	decreases	<25%
C.	increases	<75%
D.	decreases	<75%

Proposed Answer: C

Explanation (Optional):

A. Incorrect – It is not required to lower power to less than 25%.

B. Incorrect - Loss of feedwater heating reduces core inlet enthalpy which is an increase in inlet subcooling and it is not required to lower power to less than 25%.

C. Correct - Loss of feedwater heating reduces core inlet enthalpy, resulting in an increase in thermal power and a shift in thermal flux shape. By reducing Reactor thermal power by 25% of rated thermal power (i.e., approximately 500MWth) below the pre-transient value, the margin to thermal limits will improve

D. Incorrect - Loss of feedwater heating reduces core inlet enthalpy which is an increase in inlet subcooling.

Technical Reference(s): PNPS 2.4.150, pgs 2, 4 & 5.
O-RO-01-03-09, pages 33-37 of 64 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-04-09 EO 3 (As available)

Question Source: Bank # TADs ID: 5296
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: Not used

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

10 CFR Part 55 Content: 55.41 5
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	216000 K4.09	
	Importance Rating	3.3	

Knowledge of NUCLEAR BOILER INSTRUMENTATION design feature(s) and/or interlocks which provide for the following: Protection against filling the main steam lines from the feed system

Proposed Question: RO Question # 30

With the REACTOR FEED PUMP HI WATER LEVEL TRIP CUTOUT SWITCH on the C905 panel in the "ON" position, which one of the following will prevent the feedwater pumps from filling the main steam lines following a Reactor scram?

The reactor feed water pumps will trip when:

A. Either Narrow Range LI-263-100A OR B indicates +60 inches or greater.

B. When both Narrow Range Instruments LI-263-100A AND B indicate +60 inches or greater.

C. Either Feedwater Level Control Instrument LI-640-29A OR B indicates +60 inches or greater.

D. Both Feedwater Level Control Instruments LI-640-29A AND B indicate +60 inches or greater.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Level indicators (LI-640-29A/B) on panel 905 provide the feedwater pump trip level indication. Both level circuits must sense high level to trip the feedwater pumps.

B. Incorrect – Level indicators (LI-640-29A/B) on panel 905 provide the feedwater pump trip level indication.

C. Incorrect - Both level circuits must sense high level to trip the feedwater pumps.

D. Correct - Reactor vessel water level is measured by two identical, independent sensing systems. Level transmitters (LT-646A/B). The level signals are fed to two level indicators (LI-640-29A/B) on panel 905. Each level sensing analog instrument in the level sensing circuit system is equipped with a bistable device (640-44A/B) that provides a signal to trip the feedwater pumps and alarm at the main control room when extreme high water level is detected (+60"). Both level circuits must sense high level to trip the feedwater pumps.

Technical Reference(s): PNPS 2.2.96, Sect. 4.3[5] pg 14 (Attach if not previously provided)
Feedwater Control SD pgs 8 & 9.

Proposed References to be provided to applicants during examination: None

Learning Objective: RO-02-06-01, EO-3e (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201006 K5.10	
	Importance Rating	3.2	

Knowledge of ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) design feature(s) and/or interlocks which provide for the following: Withdraw error: P-Spec(Not-BWR6)

Proposed Question: RO Question # 31

Given the following:

- A Reactor Startup is in progress with reactor power in the source range;
- The current and following steps of the Rod Withdrawal Sequence and associated rod positions are as shown below:

Group	Step	Rod	Move From/To	Current Position
8	27	14 - 39	08 to 12	12
		38 - 15	08 to 12	12
		38 - 39	08 to 12	10
		14 - 15	08 to 12	08
9	28	30 - 31	08 to 12	08
		22 - 31	08 to 12	08
		30 - 23	08 to 12	08
		22 - 23	08 to 12	08

- There are NO Rod Worth Minimizer (RWM) errors currently existing;
- Control Rod 38-39 is selected for withdraw and being notch withdrawn from position 10 to position 12.

When the rod is withdrawn, the rod "double-notches" and settles at position 14.

Which one of the following is correct regarding further control rod movement?

The RWM will automatically block .....

A. ANY control rods from being inserted or withdrawn. Rod 38-39 can ONLY be repositioned after bypassing the RWM.

B. Control Rod 38-39 from further withdraw but it can be inserted back to position 12. NO other rod movement is possible unless the RWM is bypassed.

C. Control Rod 38-39 from further withdraw but can be inserted back to position 12. The remaining rods in step 27 can ALSO be inserted or withdrawn provided movement is within the limits of the step.

D. ANY control rods from being withdrawn. ALL control rods can be inserted until three insert errors are created.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The error rod can be inserted to correct the withdrawal error.

B. Correct: A withdrawal error was generated when rod 38-39 was withdrawn past it's withdraw limit. Until the error is corrected, all other rod motion is inhibited.

C. Incorrect: A withdrawal error was generated when rod 38-39 was withdrawn past it's withdraw limit. Until the error is corrected, all other rod motion is inhibited. Plausible in that the RWM does not enforce how the rods are withdrawn or inserted within the step provided the insert and withdrawal limits are not violated.

D. Incorrect: All rod movement is inhibited unless the withdraw error is corrected. Plausible in that normally, operation can continue if an insert error is made provided that there are no more than three insert errors.

Technical Reference(s): PNPS 2.2.90, Sect. 4, pgs 9-11 and page 25 (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: Not used

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	204000 K6.08	
	Importance Rating	3.5	

Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER CLEANUP SYSTEM: PCIS/NSSSS

Proposed Question: RO Question # 32

Given the following:

- The plant is at rated conditions;
- Reactor Water Cleanup is in service;

Then, a loss of 120VAC Safeguard Bus Y-3 occurs.

Which of the following RWCU valves will automatically close?

1. RWCU Inboard Isolation Valve MO-1201-2
2. RWCU Outboard Isolation Valve MO-1201-5
3. RWCU Outboard Isolation Valve MO-1201-80

A. 1 only

B. 2 only

C. 2 and 3 only

D. 1 and 3 only

Proposed Answer: A

Explanation (Optional):

A. Correct - The Group VI isolation (RWCU) utilizes a two channel, normally energized, de-energized to trip logic powered by 120 V essential service panels Y-3 and Y-4. Channel A, powered from Y-3, trips de-energize relay 16A-K26 which shuts the inboard supply valve to the RWCU system, MO-2. Channel B, powered from Y-4, trips de-energize relay 16A-K27 which shuts the outboard supply valve to the RWCU system, MO-5, and also shuts the RWCU system return valve to feedwater line "A" MO-80.

B. Incorrect - The 5 valve would close if Y-4 were lost.

C. Incorrect - This would be the response if Y-4 were lost.

D. Incorrect - Only the 2 valve will close.

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

PNPS 5.3.18, LOSS OF 120V AC SAFEGUARD BUSES Y3 AND Y31, page 3

PCIS Reference Text, page 26

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-10, EO-12p (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	272000 A1.01	
	Importance Rating	3.2	

Ability to predict and/or monitor changes in parameters associated with operating the RADIATION MONITORING SYSTEM controls including: Lights, alarms, and indications associated with normal operations

Proposed Question: RO Question # 33

With the plant operating at full power the following occur:

- Steam Jet Air Ejector Offgas monitor 1705-3A is inoperable
- The selector switch for 1705-3A has been placed in the INOP position
- The OFF GAS ISOL CH PRM SEL switch has been moved from position 2 to position 1.

Based on these switch re-alignments which one of the following will cause Annunciator, 13 MIN TIMER INITIATED, CP600R-B3 and the subsequent Offgas isolation?

A. Steam Jet Air Ejector Offgas monitor 1705-3B exceeds its Hi - Hi setpoint only.

B. Post Treatment Offgas Rad Monitor 1705-5A OR 1705-5B exceed their Hi - Hi setpoints only.

C. Steam Jet Air Ejector Offgas monitor 1705-3B exceeds its Hi - Hi setpoint or indicates downscale.

D. BOTH Post Treatment Offgas Rad Monitor 1705-5A AND 1705-5B exceed their Hi - Hi setpoints or indicate downscale.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: If the OFFGAS ISOL CH PRM SEL switch is in MON-1, then the post treat PRM (1705-5A/B) will start the 13 min. timer. The post treat PRM is measuring activity at the outlet of the charcoal vault. If the OFFGAS ISOL CH PRM SEL switch is in MON-2, then the pre treat PRM (1705-3A/B) will start the 13 min. timer. The pre treat PRM is measuring the activity at the air ejectors

B. Incorrect: Both channels must exceed their Hi-Hi trip settings.

C. Incorrect: With the selector switch in the MON 1 position, the SJAE monitor does not input into the trip circuit.

D. Correct - If the OFFGAS ISOL CH PRM SEL switch is in MON-1, then the post treat PRM (1705-5A/B) will start the 13 min. timer. If the OFFGAS ISOL CH PRM SEL switch is in MON-2, then the pre treat PRM (1705-3A/B) will start the 13 min. timer. To cause the alarm and isolation both channels of selected Rad Monitor must be upscale or both channels of selected Rad Monitor must be downscale

Technical Reference(s): ARP-CP600R, B-3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-03-02, EO-6c (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	288000 A2.04	
	Importance Rating	3.7	

Ability to (a) predict the impacts of the following on the PLANT VENTILATION SYSTEMS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: High radiation: Plant-Specific

Proposed Question: RO Question # 34

The plant is operating normally at 100% power when the following occur:

- Annunciator 904CL-B-5, REACTOR BLDG VENT RAD HI alarms
- Annunciator 904LC-A-5, REACTOR BLDG VENT RAD HI - HI alarms

Based on only these annunciators which one of the following is the status of Secondary Containment Ventilation and whether entry into EOP-4, Secondary Containment Control is required?

	Secondary Containment has:	EOP-4 entry is:
A.	automatically isolated	required
B.	automatically isolated	NOT required
C.	NOT automatically isolated	required
D.	NOT automatically isolated	NOT required

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Neither annunciator 904CL-B-5 or 904CL-A-5, REACTOR BLDG VENT RAD HI and HI HI are not tied to an automatic isolation of the Secondary Containment or SBGT start.

B. Incorrect – Neither annunciator 904CL-B-5 or 904CL-A-5, REACTOR BLDG VENT RAD HI and HI HI are not tied to an automatic isolation of the Secondary Containment or SBGT start. REACTOR BLDG VENT RAD HI alarm (panel C904LC, B5) is an entry condition for EOP-4.

C. Correct – Neither annunciator 904CL-B-5 or 904CL-A-5, REACTOR BLDG VENT RAD HI and HI HI are not tied to an automatic isolation of the Secondary Containment or SBGT start. REACTOR BLDG VENT RAD HI alarm (panel C904LC, B5) is an entry condition for EOP-4.

D. Incorrect - REACTOR BLDG VENT RAD HI alarm (panel C904LC, B5) is an entry condition for EOP-4.

Technical Reference(s): EOP-04 (Attach if not previously provided)
ARP-904CL, A-5 and B-5

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-05 EO 14c (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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	239001 K2.01	
	Importance Rating	3.2	

Knowledge of electrical power supplies to the following: Main steam isolation valve solenoids

Proposed Question: RO Question # 35

The plant is operating at 100% power when power is lost to 120V AC SAFEGUARD BUSES Y3.

Which one of the following is the effect on the Main Steam Isolation Valves (MSIVs)?

The AC solenoids for the ...

A. inboard MSIVs are de-energized and the valves close

B. outboard MSIVs are de-energized and the valves close

C. inboard MSIVs are de-energized and the valves remain open

D. outboard MSIVs are de-energized and the valves remain open

Proposed Answer: C

Explanation (Optional):

A. Incorrect - Both pilots must be de-energized to allow air to the piston top to close the valve. This prevents inadvertent closing of MSIV if one solenoid power supply is lost.

B. Incorrect - Y3 powers the inboard solenoids. Both pilots must be de-energized to allow air to the piston top to close the valve. This prevents inadvertent closing of MSIV if one solenoid power supply is lost.

C. Correct - Inboard MSIV solenoids are powered from panel D-6 for the 125 VDC solenoids, and from panel Y-3 for the 120 V solenoids. Outboard MSIV solenoids are powered from panel D-5 for the 125 VDC solenoids and from panel Y-4 for the 120 V solenoids. Loss of any one of these 4 power supplies will cause the amber logic lights on C-905 to extinguish. Loss of one power supply will not cause any MSIV to close. Loss of panels D-6 <u>and</u> Y-3 will cause the inboard MSIVs to close. Loss of panels D-5 <u>and</u> Y-4 will cause the outboard

MSIVs to close.

D. Incorrect - Y3 powers the inboard solenoids.

Technical Reference(s): PNPS 2.2.92, Att. 3 (Attach if not previously provided)
Main Steam System Description,
pg 31, Sect 14.a

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-02-08-10, EO-12j, k (As available)

Question Source: Bank # TADs ID: 3981

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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	245000 A4.02	
	Importance Rating	3.1	

Ability to manually operate and/or monitor in the control room: Generator controls

Proposed Question: RO Question # 36

The plant is operating at 20% power with Main Generator voltage control in AUTO when a faulty generator output voltage signal causes the Generator output voltage to rise to the Maximum excitation limit.

Which one of the following describes the plant response?

A. The main generator will trip initiating a turbine trip and reactor scram

B. The main generator will trip initiating a turbine trip, the reactor will NOT scram.

C. The voltage regulator will transfer automatically back to the MANUAL mode at the last operator set manual setting

D. The voltage regulator will transfer automatically back to the MANUAL mode at the last automatic setting prior to the failure.

Proposed Answer: C

Explanation (Optional):

A. Incorrect - The voltage regulator shifts to manual no generator/turbine trips or scrams occur

B. Incorrect - The voltage regulator shifts to manual no generator/turbine trips or scrams occur

C. Correct - the voltage regulator will transfer automatically back to the MANUAL mode should any one of these conditions occur:
(a) Exciter field breaker trip
(b) Main Generator field breaker trip
(c) Generator voltage unbalanced
(d) Maximum excitation limit
(e) Rectifiers overcurrent
(f) Excessive volts per cycle

D. Incorrect - The manual voltage regulator does not follow the auto regulator and will remain at its last setting

Technical Reference(s): PNPS 2.2.2, Sect. 4.2 [6], pg 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 2009 Audit #30
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

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	201001 2.2.25	
	Importance Rating	3.2	

Equipment Control: Knowledge of bases in technical specifications for limiting conditions for operations and safety limits. (CRD Hydraulic)

Proposed Question: RO Question # 37

Given the following:

- A reactor plant startup is in progress
- Reactor pressure is 750 psig
- Annunciator CRD PUMP A TRIP C905R-A5 alarms
- Two minutes later, annunciator ACCUMULATOR TROUBLE C905R-F6 also alarms
- The ACCUM Trouble light on the full core display for rod 18-41 is ON
- Rod 18-41 is at position 24

Tech Specs require that _____ (1) _____. The bases for this action is because _____ (2) _____.

A. (1) an immediate reactor scram be inserted.
(2) there is insufficient reactor pressure to fully insert the control rod should a reactor scram occur.

B. (1) an immediate reactor scram be inserted.
(2) although the rod will insert should a scram occur, normal scram times may be exceeded.

C. (1) if a charging water flow is not restored within 20 minutes an immediate manual scram shall be inserted.
(3) there is insufficient reactor pressure to fully insert the control rod should a reactor scram occur.

D. (2) if a charging water flow is not restored within 20 minutes an immediate manual scram shall be inserted.
(3) although the rod will insert should a scram occur, normal scram times may be exceeded.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: AT 750 psig there is still sufficient reactor pressure to insert the control rod.

B. Correct: Per Tech Spec 3.3. C.D, with reactor pressure less than 950, an accumulator trouble alarm on a non fully inserted control rod, an immediate manual scram is required. The bases for this action is that normal scram times may be exceeded. Per the bases of section 3.3.D, "Below 800 psig reactor pressure, the scram accumulators are necessary to scram the control rods within the required insertion times of LCO 3.3.C, "Scram Insertion Times."

C. Incorrect: Per Tech Spec 3.3. C.D, with reactor pressure less than 950, an accumulator trouble alarm on a non fully inserted control rod, an immediate manual scram is required. Plausible in that this is the required action at pressures > 950 psi in conjunction with two inop accumulators.

D. Incorrect: Per Tech Spec 3.3. C.D, with reactor pressure less than 950, an accumulator trouble alarm on a non fully inserted control rod, an immediate manual scram is required.

Technical Reference(s): T.S 3.3. C and D (Attach if not previously provided)

Tech Spec bases page 3 / 4.3-22

CRDM Reference Text, figure 25

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 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	290003 K4.01	
	Importance Rating	3.1	

Knowledge of CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following: System initiations/reconfiguration: Plant-Specific

Proposed Question: RO Question # 38

The Control Room Ventilation System is operating in the "normal" configuration. The alignment of the CRHEAF system is as follows:

- CRHEAF SUPPLY FAN A is in AUTO
- CRHEAF SUPPLY FAN B is in AUTO

While operating in this configuration, what will be the effect of a Halon discharge in the Cable Spreading Room?

A. Normal supply and recirculation fans trip, only one CRHEAF supply fan starts.

B. Normal supply and recirculation fans trip, both CRHEAF supply fans start.

C. Normal supply fans continue to run, recirculation fan trips, only one CRHEAF supply fan starts.

D. Normal supply fans trip, recirculation fan remains running, both CRHEAF supply fans start.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Both fans start

B. Correct - During discharge of the Halon into the Cable Spreading Room, the supply fans (VAC-104A and VAC-104B) and recirculation fans (VRF-101A and VRF-101B) are shut down and the Control Room environmental air system fans (VSF-103A and VSF-103B, CRHEAF SUPPLY FANS A and B) are started. This provides fresh filtered air to Control Room personnel while the normal HVAC is shut down. It also pressurizes the Control Room pursuant to FSAR Section 10.17.

C. Incorrect – Normal supply fans and recirculation fans stop

D. Incorrect - Normal supply fans and recirculation fans stop

Technical Reference(s): 2.2.46, Sect. 4.2 [6], pgs 9 & 10 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: O-RO-03-08-03 EO-5 (As available)

Question Source: Bank # TADs ID: 3166
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 8
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295037 EK1.03	
	Importance Rating	4.2	

Knowledge of the operational implications of the following concepts as they apply to SCRAM CONDITION PRESENT AND REACTOR POWER ABOVE APRM DOWNSCALE OR UNKNOWN: Boron effects on reactor power (SBLC)

Proposed Question: RO # 39

Given the following conditions:

- The plant was at rated conditions when a complete loss of normal feed resulted in a reactor scram.
- Control rods failed to insert and EOP-02, Failure to Scram, is being executed.
- RPV Pressure is being maintained between 900 and 1050 psig via SRVs
- RPV level has been intentionally lowered and is now being controlled between -150" and -125 inches (Actual)
- Standby Liquid is being injected

When will EOP-02 direct that RPV Level be restored to normal?

NOT UNTIL.....

A. Reactor Power is below the APRM downscales.

B. The Hot Shutdown Boron Weight has been injected

C. The Cold Shutdown Boron Weight has been injected

D. Reactor Power is on IRM range 7 or lower, and continuing to lower.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Reactor level cannot be restored until the Hot Shutdown Boron Weight has been injected. Plausible in that this is one of the criteria for stopping the intentional lowering of level.

B. Correct: EOP-02, Step L-7 specifies that when the Hot Shutdown Boron Weight (HSBW) has been injected than the "R" leg of EOP-02 is to be executed. This leg directs that RPV level be restored to normal via step L-23.

C. Incorrect: Level can be restored when the HSBW has been injected. Delaying the level

restoration may delay the reactor shutdown due to the imperfect boron mixing that may have occurred when level was lowered and core flow dropped.

D. Incorrect: Level can be restored when the HSBW has been injected. Plausible in that this is the EOP-02 definition of reactor shutdown.

Technical Reference(s): EOP-02, Failure to Scram. (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295005 AK1.03	
	Importance Rating	3.5	

Knowledge of the operational implications of the following concepts as they apply to MAIN TURBINE GENERATOR TRIP: Pressure effects on reactor level

Proposed Question: RO # 40

The reactor is at steady-state, rated power conditions at the beginning of the fuel cycle (BOC). All systems are operable.

Then, a spurious Main Turbine trip occurs. Reactor water level will _____.

(Assume NO operator action)

A. initially shrink until HPCI/RCIC automatically initiates to restore reactor level.

B. initially swell until feedwater control automatically stabilizes level at the original level.

C. shrink and swell continuously due to SRV cycling until feedwater control automatically stabilizes level at the original level.

D. shrink and then swell following momentary SRV cycling and then continue to rise until the feedwater pumps trip on high RPV level due to the feed reg valve leakage.

Proposed Answer: D

Explanation (Optional):

A. Incorrect - A main turbine trip would not result in a level decrease to the extent of initiation of HPCI or RCIC systems. This is characteristic for Loss of Vacuum or MSIV Closure events.

B. Incorrect – Level will shrink due to the reactor pressure rise resulting from the turbine trip. The level will not stabilize at the original level without operators taking manual control of the feedwater

C. Incorrect – SRV cycling is indicative of a main turbine trip without bypass capability.

D. Correct - level will shrink due to the reactor pressure rise resulting from the turbine trip, as well as due to the loss of void production and core flow

sweeping out the pre-scram voids. Feedwater restores level until the feedwater pumps trip on High RPV level unless the operators take manual control of the system. PNPS 2.1.6 step [5] requires the operators to take manual control of the Feed Reg Valves and Trip the feedwater pumps as required to maintain level.

Technical Reference(s): PNPS 2.1.6 step [5]b. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS4623

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

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

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

Proposed Question: RO # 41

Given the following:

- The plant is at rated conditions
- RCIC is in service for Quarterly Operability Testing
- "A" CRD Pump is in service
- RBCCW valve MO-4085A, RBCCW Loop "A" Non-Essential Loop Inlet Valve fails full CLOSED

Assuming no operator action is taken, which one of the following will occur?

A. RCIC will isolate on high area temperature.

B. "A" CRD pump will seize due to loss of cooling to its bearings.

C. BOTH "A" AND "B" Recirc MG sets will trip on high oil temperature.

D. RWCU will isolate on high Non Regenerative Heat Exchanger outlet temperature.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: RCIC will not isolate. Plausible in that RCIC area coolers are cooled by "A" RBCCW. However these coolers are essential loads that were not isolated by the valve closure.

B. Incorrect: Both CRD pumps are cooled by the "B" RBCCW loop.

C. Correct: The MG set oil coolers for BOTH Recirc MG sets are non-essential loads off the "A" loop. The MG Sets will trip when lube oil temperatures reach 210 degrees.

D. Incorrect: RWCU is cooled by the "B" RBCCW loop.

Technical Reference(s): PNPS 2.2.30, RBCCW page 9 (Attach if not previously provided)

PNPS 2.2.84, REACTOR RECIRCULATION SYSTEM, page 19

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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295006 AK2.06	
	Importance Rating	4.2	

Knowledge of the interrelations between SCRAM and the following: Reactor Power

Proposed Question: RO # 42

Given the following:

- A Reactor Startup is in progress
- The reactor mode switch is in the STARTUP position, about to be transferred to RUN
- The Main Turbine is on the turning gear with all stop valves open
- Reactor pressure is 940 psig
- Reactor power is 25 on Range 10 of the IRMs

Then.....

- An inadvertent HPCI injection occurs
- The reactor automatically scrams in response to the transient

What caused the reactor to scram?

A. Reactor power exceeded the IRM Hi-Hi setpoint

B. Reactor power exceeded the APRM Hi-Hi setpoint

C. Water level exceeded the high Main Turbine trip setpoint

D. Water level exceeded the high MSIV Closure I isolation setpoint

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The IRMs will generate a scram when they exceed 120/125ths of scale. A reading of 100 on range 10 equates to ~ 32% power. Therefore the IRMs will not reach their high-high setpoint until power is somewhat greater than 32%. The APRM Hi-Hi will occur at 15% before power can reach that level.

B. Correct: With the mode switch in Startup, the APRM set down setpoints are in effect. This will cause a scram when power exceeds 15%. Therefore the reactor will scram on

APRM Hi-Hi.

C. Incorrect: Although the turbine may trip when level reaches +45 inches, the main turbine trips are bypassed due to the low 1st stage pressure in this condition (on the turning gear)

D. Incorrect: HPCI will trip at +45 inches. The Group I high level setpoint is + 55 inches. Additionally, reactor pressure is too high to cause an isolation even if level reached + 55.

Technical Reference(s): TS Table 3.1.1 and associated notes.
IRM Reference Text, page 10 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295016 AK2.02	
	Importance Rating	4.0	

Knowledge of the interrelations between CONTROL ROOM ABANDONMENT and the following: Local control stations: Plant-Specific

Proposed Question: RO # 43

The plant was operating at rated power.

Following a loss of feedwater and a subsequent control room evacuation, HPCI automatically initiated at -46 inches. System control is now taken at the alternate shutdown panel (ASP) by aligning all HPCI LOCAL/REMOTE control switches for local operation.

How will HPCI respond if RPV level continues to rise above +45"?

A. HPCI will trip

B. HPCI will continue to inject.

C. HPCI will isolate.

D. HPCI will be operating on minimum flow.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – HPCI will continue to inject because trips are bypassed with control at the ASP

B. Correct – IAW PNPS 2.4.143, APP.A, Transfer of HPCI control to the ASP along with breaker manipulations performed in Appendix F bypasses all HPCI initiation, trip, and interlock functions except the Turbine overspeed trip

C. Incorrect – HPCI will continue to inject

D. Incorrect – HPCI will continue to inject

Technical Reference(s): PNPS 2.4.143 App. A Step 2.0[2] (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # TAD ID - 2369

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295028 EK2.02	
	Importance Rating	3.2	

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

Proposed Question: RO # 44

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) **Maximum Drywell Internal Temperature**	(2) **Maximum SRV Solenoid Temperature**
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: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 9

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 AK3.02	
	Importance Rating	3.6	

Knowledge of the reasons for the following responses as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID DISTURBANCES: Actions contained in abnormal operating procedure for voltage and grid disturbances.

Proposed Question: RO # 45

Given the following:

- The plant is at rated conditions;
- A degraded voltage condition exists
- PNPS, 2.4.144, DEGRADED VOLTAGE has been entered

PNPS 2.4.144 states, DO NOT operate Core Spray or RHR Pump(s) with the EDG in parallel with the Startup or Unit Aux Transformers.

The reason for this is to _____.

A. ensure EDG loading is consistent with the engineering analysis.

B. prevent a reverse power trip of the EDG.

C. prevent an overvoltage condition from occurring on the EDG.

D. ensure that the EDG will trip with an overcurrent condition while in the Isochronous mode.

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW PNPS 2.4.144, Step 4.0[6]

B. Incorrect – total EDG loading is the concern

C. Incorrect – voltage will be approximately at rated however, current will be lower than normal

D. Incorrect- total EDG loading is the concern

Technical Reference(s): PNPS 2.1.144, Step 4.0[6] (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 EK3.05	
	Importance Rating	3.9	

Knowledge of the reasons for the following responses as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE: Reactor SCRAM

Proposed Question: RO # 46

Given the following:

- The reactor is at 10% power
- EOP-03, Primary Containment Control has just been entered due to rising torus water temperature.

Based on the above EOP-03 requires that EOP-01, RPV Control, be entered before torus water temperature exceeds the _____ (1) _____. The basis for this action is to ____ (2) _____?

A. (1) Heat Capacity Temperature Limit
(2) ensure that, if possible, the reactor is scrammed and RPV pressure and level control are established prior to commencing an Emergency Depressurization

B. (1) Heat Capacity Temperature Limit
(2) ensure that, if possible, the reactor is scrammed in order to limit additional heat input to the torus to prevent incomplete steam condensation following a blowdown.

C. (1) Boron Injection Initiation Temperature Limit
(2) ensure that, if possible, the reactor is scrammed and shutdown by control rod insertion before the requirement for boron injection is reached. If rods fail to insert, sufficient boron can be injected before torus water temperature exceeds the Heat Capacity Temperature Limit.

D. (1) Boron Injection Initiation Temperature Limit
(2) ensure that, if possible, the reactor is scrammed and shutdown by control rod insertion before the requirement for boron injection is reached. If rods fail to insert, sufficient boron can be injected before torus water temperature exceeds low pressure ECCS NPSH requirements.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Per EOP-03, step TT-9, EOP-01 is entered before the torus water temperature exceeds the Boron Injection Initiation Temperature limit. Emergency Depressurization is required when the HCTL is exceeded which is much higher than the BIIT limit.

B. Incorrect: Per EOP-03, step TT-9, EOP-01 is entered before the torus water temperature exceeds the Boron Injection Initiation Temperature limit.

C. Correct: Per EOP-03, step TT-9, EOP-01 is entered before the torus water temperature exceeds the Boron Injection Initiation Temperature limit. Per the EOP-03 LP, Entering EOP-01 at Step R-1 assures that, if possible, the reactor is scrammed and shutdown by control rod insertion before the requirement for boron injection is reached. Conditions requiring entry into EOP-03 do not necessarily require entry into the RPV Control guideline. Therefore, a scram may not have yet been initiated. Also at 10% power the BIIT is 121 degrees. This is, the temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight (HSBW) before torus water temperature exceeds the Heat Capacity Temperature Limit (HCTL).

D. Incorrect: The BIIT curve It is the greater of either the highest torus water temperature at which initiation of boron injection will permit injection of the Hot Shutdown Boron Weight (HSBW) before torus water temperature exceeds the Heat Capacity Temperature Limit (HCTL) or the torus water temperature at which a reactor scram is required by Technical Specifications (110ºF). At 10% power the temperature the HCTL is limiting.

Technical Reference(s): EOP-03, step TT-9 (Attach if not previously provided)

EOP-03 LP Section VIIII.B.10

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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295024 EK3.07	
	Importance Rating	3.5	

Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL PRESSURE: Drywell venting

Proposed Question: RO # 47

The plant was operating at rated power when a LOCA occurred. Conditions have deteriorated to the point where the primary containment pressure limit has been reached.

Torus Water Level is pegged high at greater than 300 inches.

The EOPs require emergency venting through the

A. torus vents to ensure that some scrubbing action of the discharge stream will occur.

B. torus vents because the drywell vents provide an unfiltered release path

C. drywell vents to ensure an elevated release point.

D. drywell vents because the torus vents are covered.

Proposed Answer: D

Explanation (Optional):

A. Incorrect – with torus level above 300 inches, the torus vents are covered

B. Incorrect - with torus level above 300 inches, the torus vents are covered

C. Incorrect – although this an elevated release point, the reason drywell vents are used is because the torus vents are covered

D. Correct – EOP-03 LP Section X.B.7.d., discussion of EOP step P-7. If the torus water level is above 300", And below 77', the operator is directed to vent through the drywell vents because the torus vents are submerged.

Technical Reference(s): EOP-03 LP Section X.B.7.d., discussion of EOP step P-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 X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295023 AA1.04	
	Importance Rating	3.4	

Ability to operate and/or monitor the following as they apply to REFUELING ACCIDENTS: Radiation monitoring equipment.

Proposed Question: RO # 48

The plant is shutting down for refueling. The Refuel Floor Ventilation Exhaust monitors have not yet been setup to their refueling values.

The "A" Channel Refuel Floor Ventilation Exhaust (RFVE) Radiation Monitor is failed downscale. The other channels are operable.

A new fuel movement accident on the refuel floor results in the following

- At T=0 minutes:
    - "B" Channel RFVE reading 13 mR/hr
    - "C" Channel RFVE reading 14 mR/hr
    - "D" Channel RFVE reading 17 mR/hr
- At T=1minute: "C" Channel RFVE fails downscale
- At T=2 minutes: "B" Channel RFVE reading increased to 16 mR/hr

Which one of the following describes if/when a Reactor Building (RB) Isolation occurred?

An RB Isolation _____.

A. has not occurred.

B. occurred BEFORE the "C" RFVE channel failed downscale.

C. occurred WHEN the "C" RFVE channel failed downscale.

D. occurred WHEN the "B" RFVE increased to 16 mR/hr.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – an isolation occurred when the "C" channel failed downscale

B. Incorrect – the 1 out of 2 twice logic has not been met

C. Correct IAW PRM Reference text – page 19 – component descriptions - To eliminate the possibility of a faulty detector unnecessarily isolating the Reactor building and primary containment atmosphere control systems but still provide protection should all four detectors fail, the protection logic will initiate the isolations under any of the following conditions: One rad monitor in each channel senses a high radiation condition, OR;
All four rad monitors are downscale, OR;
Both rad monitors in one channel are downscale, while one of the rad monitors in the other channel senses a refuel floor high radiation condition. (16 mR/hr)

D. Incorrect – the isolation occurred prior to that time

Technical Reference(s): PRM Reference text – page 19 (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:

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 AA1.06	
	Importance Rating	3.0	

Ability to operate and / or monitor the following as they apply to PLANT FIRE ON SITE: Fire alarm

Proposed Question: RO # 49

Given the following:

- The plant is at rated conditions with a normal configuration on electrical distribution system
- A fire alarm on Panel C-7, TRANSFORMER HEADER TROUBLE, C7R-B1, occurs.
- The alarm is determined to be due to a trip of a Heat Actuated Device associated with the Startup Transformer.
- No other alarm signals are present for the Startup Transformer

IAW PNPS 5.5.2, Special Fire Procedure, verify that …

A. deluge has automatically initiated.

B. deluge automatically initiates if there is a concurrent Startup Transformer lockout condition present.

C. deluge automatically initiates if there is a concurrent Startup Transformer Sudden Pressure condition present.

D. the Startup Transformer has automatically locked out and manually initiate deluge.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Deluge will not initiate unless there is a concurrent Startup Transformer Lockout signal. Plausible in that deluge will initiate for the Shutdown or the Unit Aux transformers if only the HAD actuates.

B. Correct: Automatic Deluge initiation requires concurrent signals of Startup transformer lockout and HAD actuation.

C. Correct – Incorrect. The sudden pressure condition /alarm will not result in deludge actuation.

D. Incorrect – HAD actuation will not cause a startup transformer lockout.

Technical Reference(s): PNPS 2.2.26, DELUGE, SPRINKLER, AND SPRAY SYSTEMS Page 13 ARP C7R-B1 (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 4

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA1.03	
	Importance Rating	4.4	

Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER : Systems necessary to assure safe plant shutdown

Proposed Question: RO # 50

The plant was operating at rated power. Given the following conditions:

- A Loss of Off-Site Power has occurred.
- An electrical fault on bus A5 results in a bus A5 lockout.
- Drywell pressure rises to 25 psig.
- Reactor pressure has lowered to 45 psig.

Which of the following describes pumps that will be injecting into the RPV?

A. HPCI, 2 RHR Pumps and 1 Core Spray Pump

B. 1 RHR Pump and 1 Core Spray Pump

C. RCIC, 1 RHR Pump and 1 Core Spray Pump

D. 2 RHR Pumps and 1 Core Spray Pump

Proposed Answer: D

Explanation (Optional):

A. Incorrect – HPCI isolates at RPV pressure of 80 psig – HPCI reference text page 44

B. Incorrect – 2 RHR pumps would be running – A & C are powered from A5 and would not be running – RHR reference text page 23

C. Incorrect - RCIC isolates at RPV pressure of 50 psig – RCIC reference text page 8. 2 RHR pumps would be running – A & C are powered from A5 and would not be running – RHR reference text page 23

D. Correct – The loss of A5 results in only 2 RHR pumps in service from A6 and one core spray pump running powered from A6 – 4160 V reference text, att.1 table

Technical Reference(s): HPCI, RCIC, RHR, CS, 4160V Reference Texts (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Pilgrim NRC 2003

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	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 # 51

The plant is operating at rated power when Annunciator CP600R, MAIN STACK RAD HI-HI alarms.

Which one of the following describes a source that would cause the above condition?

A. A Recirc Pump seal failure.

B. Main Condenser Offgas high radiation levels.

C. HPCI steam leak in the HPCI Turbine Area

D. A Radwaste effluent leak into the equipment drain system.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – this would result in rising DW pressure and temperature. Containment rad levels would rise

B. Correct – IAW PRM reference text, page 20, Main stack effluent is the combined effluent from:
   - Main condenser off-gas system
   - Primary containment atmosphere control (in purge or inerting mode)
   - Control room HVAC
   - Standby Gas Treatment discharge

C. Incorrect – HPCI steam leak would be monitored by the Reactor Building Ventilation Exhaust PRMs

D. Incorrect – this system is not tied to the Main Stack PRM system

Technical Reference(s): PRM Reference Text page 20 (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 11

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295019 AA2.02	
	Importance Rating	3.6	

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF INSTRUMENT AIR : Status of safety-related instrument air system loads (see AK2.1 - AK2.19)

Proposed Question: RO # 52

Instrument Air Pressure is slowly being lost. The K-117 air compressor fails to start and header pressure continues to lower.

RBCCW system temperature __ (1) __ and RBCCW surge tank level __ (2) __.

A. (1) Increases
(2) Increases

B. (1) Increases
(2) Decreases

C. (1) Decreases
(2) Decreases

D. (1) Decreases
(2) Increases

Proposed Answer: D

Explanation (Optional):

A. Incorrect – Temperature decreases

B. Incorrect – Temperature decreases and level increases

C. Incorrect – level increases

D. Correct – IAW PNPS 5.3.8 Att.1 , RBCCW TCV (bypasses heat exchanger) fails closed giving maximum cooling to RBCCW, RBCCW surge tank LCV fails open, raising level

Technical Reference(s): PNPS 5.3.8 Att.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # NRC 2002

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2002

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295031 EA2.04	
	Importance Rating	4.6	

Ability to determine and/or interpret the following as they apply to REACTOR LOW WATER LEVEL : Adequate Core Cooling

Proposed Question: RO # 53

Given the following:

- A LOCA has occurred.
- Actual RPV Water level is -170 inches and steady

Based on the above which one of the following will satisfy adequate core cooling requirements IAW EOP-01? Assume there are no other injection sources available.

A. Core Spray "A" injecting at 4000 gpm
RHR Pump "C" injecting at 3000 gpm

B. Core Spray "A" injecting at 2000 gpm
Core Spray "B" injecting at 3000 gpm

C. RHR Pump "C" injecting at 4000 gpm
Core Spray "B" injecting at 3000 gpm

D. Core Spray "B" injecting at 3000 gpm
All four LPCI pump injecting with a total flow of 16,000 gpm

Proposed Answer: A

Explanation (Optional):

A. Correct: Spray cooling is satisfied when RPV level is ≥ -175 inches and at least one core spray pump is injecting at ≥ 3600 gpm.

B. Incorrect: Neither core spray subsystem is ≥ 3600 gpm. Plausible in that when combined, total flow is ≥ 3600 gpm.

C. Incorrect: With level less than -150 inches, core cooling is satisfied by spray cooling with a spray flow rate from one system at ≥ 3600 gpm. Plausible in that the RHR pump is greater than the required spray flow rate.

D. Incorrect: With level less than -150 inches, core cooling is satisfied by spray cooling with a spray flow rate from one system at ≥ 3600 gpm. Plausible in that LPCI is injecting at maximum flow rate.

Technical Reference(s): EOP-01 Reference text – descriptions of EOP steps L-15 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295025 2.4.18	
	Importance Rating	3.3	

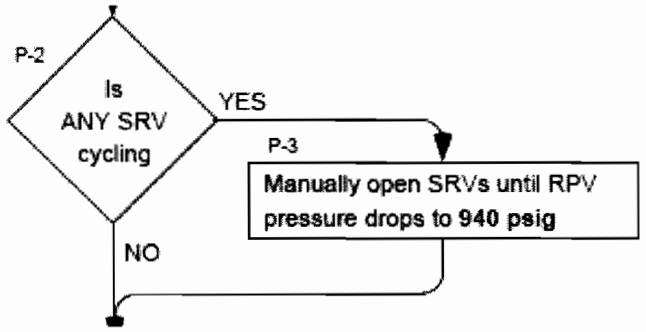
Emergency Procedures / Plan: Knowledge of the specific bases for EOPs. (High reactor pressure)

Proposed Question: RO # 54

Given the following:

- ATWS conditions exist
- EOP-02, Failure to Scram, is being executed
- All three main turbine bypass valves are open
- SRVs are lifting and pressure is cycling between ~ 1060 and 1120 psig

EOP-02, Step P-3 shown below directs that SRVs are to be manually opened and pressure lowered to 940 psig. Which one of the following describes the bases for this step?



This step is intended to lower RPV pressure in order to

A. Minimize the SRV cycling and allow the scram to be reset

B. Preserve drywell pneumatic supply by terminating the SRV cycling

C. Minimize the SRV cycling and maximize heat rejection to the main condenser

D. Provide margin to the SRV lift point but not so low as to exceed the low pressure high power safety limit

Proposed Answer: C

Explanation (Optional):

A. Incorrect – This step is intended to reduce SRV cycling and maximize heat rejection to the main condenser. Step P-5 directs that pressure be controlled < 1060 psig which allows the scram to be reset.

B. Incorrect: Although preserving drywell pneumatics is an issue if the continuous drywell pneumatic supply is lost, pneumatics are not used when the SRVs are cycling on high pressure.

C. Correct: This step is intended to reduce SRV cycling and maximize heat rejection to the main condenser. Lowering pressure further will result in bypass valve closing and reducing the amount of heat going to the main condenser.

D. Incorrect – Maintaining margin to the safety limit is not the bases of this step. Plausible in that safety limits may be exceeded during ATWS events.

Technical Reference(s): EOP-02 LP, Step P-3 description, page 42 (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295021 2.1.7	
	Importance Rating	4.4	

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

KA Match Justification – with the reactor in shutdown cooling, reactor behavior is not a concern. This question does involve evaluating the plant and assessing when to take appropriate actions based on operating characteristics

Proposed Question: RO # 55

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 _____.

A. maintained below the Group I isolation setpoint to ensure that Bypass Valves are available for use in the event that the plant becomes pressurized.

B. maintained below the HPCI Hi Level trip point to ensure that HPCI is available for use in the event that the plant becomes pressurized.

C. raised above +60 inches to promote natural circulation.

D. raised above +60 inches in preparation for initiating cooling by feed and bleed.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – must be maintained > +60 inches

B. Incorrect – must be maintained > +60 inches

C. 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: (As available)

Question Source: Bank # NRC 2003

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AK3.03	
	Importance Rating	3.1	

Knowledge of the reasons for the following responses as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :Reactor scram

Proposed Question: RO # 56

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 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, AND why?

A. Manually trip 4KV bus A4 at panel C3. This will de-energize the "B" Recirc MG set.

B. Locally trip the "B" Recirc MG Drive motor breaker at 4KV bus A4. The Recirc MG set is seized due to a loss of lube oil.

C. Locally start the "B" Recirc System DC Emergency Bearing Oil Pump at 250 VDC panel D9. The Recirc MG set is seized due to a loss of lube oil.

D. Manually scram the reactor. Scramming the reactor will cause the turbine to trip, resulting in a loss of 4KV bus A4, de-energizing the "B" Recirc MG set

Proposed Answer: D

Explanation (Optional):

A. 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.

B. 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.

C. 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.

D. 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.

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 #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295030 EK1.01	
	Importance Rating	3.8	

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

Proposed Question: RO # 57

EOP-03, Primary Containment Control, directs that Emergency RPV Depressurization be performed if Torus water level cannot be maintained above 90 inches.

What is the concern if emergency depressurization is not performed at that point in the EOPs?

A. Suppression pool temperature indication becomes invalid.

B. Condensation of steam from the SRVs cannot be assured.

C. Condensation of steam from the drywell to torus cannot be assured.

D. Vortexing at the suction ECCS pumps can begin and result in air binding of the Pumps and loss of all ECCS.

Proposed Answer: C

Explanation (Optional):

A. Incorrect – Suppression pool temperature is still valid at this level

B. Incorrect: - SRV T- quenchers begin to be become uncovered at ~ 50 inches.

C. Correct – IAW EOP-3 reference text discussion of steps TL-12 thru 15 - Emergency RPV depressurization is required at this point. Depressurizing the RPV before torus water level reaches 90 in. will help ensure that the pressure suppression feature of the torus is maintained. This is the elevation corresponding to the bottom of the downcomer vent openings.

D. Incorrect – Vortexing is an issue at torus levels from 30 -50 inches per EOP-11 graph 15

Technical Reference(s): EOP-3 reference text discussion of (Attach if not previously provided) steps TL-15

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 3323

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295001 2.2.37	
	Importance Rating	3.6	

Equipment Control: Ability to determine operability and/or availability of safety related | Equipment (Partial or Complete Loss of Forced Core Flow).

Proposed Question: RO # 58

The plant was operating at rated power

The following parameter changes are noted:

- Net MWe lowers by 55 MWe
- The Turbine Control Valves throttle closed
- Reactor pressure lowers 12 psig
- Core Plate d/p lowers 2 psid
- Total Core Flow rises approximately 1.5 Mlbm/hr

These parameter changes are indicative of:

A. EPR failure

B. Jet Pump Failure

C. An SRV failing open

D. An upscale failure of a recirculation flow controller

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Reactor pressure would be higher

B. Correct – IAW PNPS 2.4.23 – these are indications of a Jet pump failure

C. Incorrect – Reactor pressure would remain relatively stable

D. Incorrect – Power would increase

Technical Reference(s): PNPS 2.4.23 discussion (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 2048

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295029 EK1.01	
	Importance Rating	3.4	

Knowledge of the operational implications of the following concepts as they apply to HIGH SUPPRESSION POOL WATER LEVEL : Containment integrity

Proposed Question: RO # 59

A Loss of Coolant Accident inside the Drywell is in progress.

Which one of the following failures or conditions could result in exceeding the NEGATIVE design pressure rating of the containment?

A. Torus to drywell vacuum breaker failing open.

B. Torus level rising to 190 inches with drywell sprays in service.

C. Torus level rising to 175 inches with torus sprays in service.

D. SRV tailpipe vacuum breaker failing closed.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – This event would challenge the over pressure rating of the containment as steam would bypass the suppression pool.

B. Correct: At 180 inches the Torus to Drywell Vacuum Breakers begin to be covered. With drywell sprays in service, the vacuum breakers would be unable to relieve back to the drywell, resulting in the drywell going negative in pressure. EOP-03, step TL-10 directs that drywell sprays be secured at this level

C. Incorrect – 175 inches is the point at which the SRVTPLL becomes limiting. If the SRVTPLL failed, a potential loss of pressure suppression might occur which is a high pressure challenge to the containment.

D. Incorrect – A SRV Tailpipe vacuum breaker failing closed would result in a vacuum drag of water up the tailpipe. Subsequent SRV lifts would result in large hydro dynamic forces on the tailpipe with possible failure. At most this would result in a high pressure condition if the tail pipe failed.

Technical Reference(s): EOP-03 Discussion of Step TL-8, DS-2 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # LOR Exam Bank # 35

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032 EK2.01	
	Importance Rating	3.5	

Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following: Area/room coolers

Proposed Question: RO # 60

Which one of the following describes the operation of the Reactor Building Quadrant Area Coolers?

RCIC Quadrant area temperature is 95 degrees and rising.

Given this temperature and with BOTH Area Cooler control switches positioned to ____

A. run, BOTH Area Coolers are operating

B. test, the "A" Area Cooler will start and the "B" Area Cooler will start if area temperature continues to rise

C. run, NO Area Coolers will be in operation

D. test, BOTH Area Coolers will be in operation

Proposed Answer: D

Explanation (Optional):

A. Incorrect – in run, only the "A" unit would be in operation

B. Incorrect – in test, they would both be operating

C. Incorrect – The A unit would be running due to the high temperature. The B unit would start at ~ 103 degrees.

D. Correct – PNPS 2.2.48 Note at step 7.1 - All CRD, RCIC, or RHR area coolers can be run continuously, regardless of quadrant ambient temperatures, by placing the respective control switch on Panels C61, C61A or B to "TEST".

Step 4.2 - Train "A" or "C" fan-coil unit(s) in any quadrant will automatically start at approximately 93°F when control switches are in "RUN". If additional cooling is

required, Train "B" or "D" of any quadrant will start cycling at approximately 103°F.

Technical Reference(s): PNPS 2.2.48 step 4.2 and 7.1 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295015 AK3.01	
	Importance Rating	3.4	

Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM : Bypassing rod insertion blocks

Proposed Question: RO # 61

The plant was at rated power when an event occurred and a scram was required. Given the following:

- Panel C905 Rod Display Group Scram Logic White Lights are extinguished
- Panel C905 Rod Display Blue Scram Lights are lit
- Annunciator C905-F1 "SPVAH PRESSURE LO" lit
- Reactor power is 20%
- The Immediate Actions of PNPS 5.3.23, Incomplete Scram have been completed

Which one of the following describes the status of the scram and actions required to insert control rods IAW PNPS 5.3.23, Incomplete Scram?

A. A hydraulic ATWS has occurred; de-energize the scram solenoids to allow rod insertion using the scram timing test switches.

B. A hydraulic ATWS has occurred; bypass the RWM to permit RPR insertion using the Reactor Manual Control System.

C. An electrical ATWS has occurred; de-energize the scram solenoids to allow rod insertion using the scram timing test switches.

D. An electrical ATWS has occurred; bypass the RWM to permit RPR insertion using the Reactor Manual Control System.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – The scram solenoids are already de-energized.

B. Correct: A hydraulic ATWS has occurred. The Group Scram and blue lights indicate that that RPS has tripped and the scram valves have re-positioned. In order to insert the rods manually the RWM must be bypassed due to the insert block.

C. Incorrect – An electrical ATWS has not occurred.

D. Incorrect – An electrical ATWS has not occurred.

Technical Reference(s): PNPS 5.3.23 Section 3.0 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295033 EA1.08	
	Importance Rating	3.6	

Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS : Control room ventilation

Proposed Question: RO # 62

The following high radiation annunciators have alarmed:

- REACTOR BLDG VENT RAD HI (904LC-B5)
- REACTOR BLDG VENT RAD HI-HI (904LC-A5)
- REACTOR BLDG RAD HI (904LC-A7) due to both 23' Reactor Building ARMs going off scale high.
- CONTROL ROOM AIR INLET RAD HI (904LC-D6)
- CONTROL ROOM RAD HI (904LC-B7)

Based on the above:

(1) What effect (if any) will these alarms have on the Main Control Room HVAC system

AND

(2) what (if any) actions are required by control room personnel?

A. (1) The normal system suction automatically isolates and air is circulated through the high efficiency filtration system.
(2) None

B. (1) There is no automatic response to these alarms
(2) Manually initiate one train of the high efficiency filtration system.

C. (1) The normal system suction isolates and the other recirculation fan will automatically start if the C/S is in "STANDBY".
(2) None

D. (1) There is no automatic response to these alarm,
(2) Manually secure the normal system suction and place both recirculation fans in "RUN".

Proposed Answer: B

Explanation (Optional):

A. Incorrect - The system does not respond to high intake radiation the high efficiency filtration system must be manually started.

B. Correct – The Control Room must manually initiate high efficiency filtration of the outside air supplied to the Control Room. Initiation of either of the filtration fans (VSF-103A or VSF-103B, CRHEAF SUPPLY FAN A or B) closes a damper in the normal outside air intake duct and opens the inlet filtration system damper.

C. Incorrect - The system does not respond to high intake radiation the suction does not isolate, the standby fan starts on low flow if the C/S is in "STANDBY".

D. Incorrect - The high efficiency filtration system must be manually started there is no direction or benefit of starting both recirculation fans particularly with their suction damper closed.

Technical Reference(s): ARP CONTROL ROOM RAD HI (904LC-B7) (Attach if not previously provided)

LP O-NL Control Room Ventilation pages 84 and 85

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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295013 AA2.01	
	Importance Rating	3.8	

Ability to determine and/or interpret the following as they apply to HIGH SUPPRESSION POOL TEMPERATURE :Suppression pool temperature

Proposed Question: RO # 63

Given the following conditions:

- The plant is operating at 90% power.
- HPCI is being tested.
- The 'A' Loop of RHR is in Torus Cooling.
- Torus temperature is 86 degrees F and continuing to rise.

IAW PNPS Technical Specifications, if torus temperature reaches 90 degrees F, you will be required to:

A. terminate HPCI testing.

B. immediately scram the reactor.

C. immediately commence a plant shutdown.

D. begin continuously monitoring torus temperature and logging it every 5 minutes.

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW TS 3.7.A.1.f. If the suppression pool bulk temperature exceeds the limits of Specification 3.7.A.1 .d (≤90°F), RCIC, HPCI or ADS testing shall be terminated and suppression pool cooling shall be initiated.

B. Incorrect – not required until >110 degrees

C. Incorrect – shutdown is not required

D. Incorrect – temperature is logged prior to starting testing

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

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 5

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #		
	Group #		
	K/A #	295009 2.1.20	
	Importance Rating	4.6	

Conduct of Operations: Ability to interpret and execute procedure steps. (Low Reactor Water Level)

Proposed Question: RO # 64

Given the following:

- The plant is at rated conditions
- The output of the "B" Feedwater M/A Station fails to zero
- When RPV level begins to lower, the "B" Feedwater M/A Station is placed in manual but the output remains at zero.
- Recirc Flow is reduced to 43 Mlbm/hr and power stabilizes at ~ 75%
- RPV level is at 16 inches and slowly lowering.

Which one of the following is required by PNPS 2.4.49, Feedwater Malfunctions?

A. Scram the reactor and enter PNPS 2.1.6 Reactor Scram

B. Trip one Recirc pump to lower power further and enter PNPS 2.4.17, RECIRC PUMP TRIP.

C. Reduce Recirc Pump speed to minimum and enter PNPS 2.4.165, REACTOR CORE INSTABILITY.

D. Open the Startup Feedwater Regulating Valve and increase feed flow IAW PNPS 2.2.96 CONDENSATE AND FEEDWATER SYSTEM.

Proposed Answer: A

Explanation (Optional):

A. Correct: Per the immediate actions of PNPS 2.4.49, if RPV level is approaching the low RPV level Scram setpoint and unable to reverse the lowering trend then the operator is directed to insert a manual scram. Power has already been lowered to 43 Mlbm/hr so flow cannot be lowered any further. Inserting the RPR will not lower power fast enough. Opening the S/U Feed Reg valve is not allowed if the feedwater heaters are in service.

Since power was initially 100%, they must be in service.

B. Incorrect: This action is not authorized by PNPS 2.4.49. Plausible in that it would reduce power quickly.

C. Incorrect: This action is not authorized by PNPS 2.4.49. Plausible in that it would reduce power quickly.

D. Incorrect: Although this action is addressed in PNPS 2.4.49, it is only allowed if the feedwater heaters are not in service. Since power was initially 100%, they must be in service.

Technical Reference(s): PNPS 2.4.49, Feedwater Malfunctions (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295022 AA2.02	
	Importance Rating	3.3	

Ability to determine and/or interpret the following as they apply to LOSS OF CRD PUMPS:
CRD system status

Proposed Question: RO # 65

The plant is at rated power when the in-service CRD pump trips.

This would result in which of the following?

(1) heatup of CRD mechanisms
(2) Recirc pump seal pressures will equalize
(3) closing of the CRD Flow Control valve
(4) gradual depressurization of the HCU scram accumulators

A. (1) and (2)

B. (1) and (4)

C. (2) and (3)

D. (3) and (4)

Proposed Answer: B

Explanation (Optional):

A. Incorrect – Seal pressures are controlled by the status of the seals and by controlling seal staging leak off flow. Plausible in that CRD provides seal purge via the #1 seal cavity, which operates at the higher of the two seal pressures.

B. Correct – IAW PNPS 2.4.4 discussion section 5.0[3]

C. Incorrect – Seal pressures are controlled by the status of the seals and by controlling seal staging leak off flow. Plausible in that CRD provides seal purge via the #1 seal cavity, which operates at the higher of the two seal pressures. Also the flow control valve would open in an attempt to maintain system flow

D. Incorrect – The flow control valve would open in an attempt to maintain system flow

Technical Reference(s): PNPS 2.4.4 discussion section 5.0[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 # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.2	
	Importance Rating	4.1	

Conduct of Operations: Knowledge of operator responsibilities during all modes of plant operation.

Proposed Question: RO # 66

Given the following:

- The plant has been operating at 2028 MWt for the past week.
- A complete loss of the process computer now occurs.
- Operators perform a manual heat balance as directed by PNPS 2.1.10 COMPUTER DATA AND ALARMS
- The heat balance indicates that the reactor is operating at 2040 MWt

Which one of the following is correct as prescribed in PNPS 2.1.10 COMPUTER DATA AND ALARMS?

A power reduction is……

A. currently required to maintain thermal power less than the licensed power level of 2028 MWt.

B. not currently required. Provided that future heat balances do not exceed 2040 MWt a power reduction will not be required.

C. not currently required. Provided that future heat balances do not exceed 2055 MWt a power reduction will not be required.

D. currently required to lower thermal power to less than 1998 MWt due to loss of the AMAG computer.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: A power reduction is not required. As discussed in PNPS 2.1.10, a manual heat balance may be up to 3% off from that calculated by the Process computer.

B. Correct: As discussed in 2.1.10, the result of the heat balance is known to be up to 3% high. The initial heat balance done following the computer loss considered to be equal to the preloss computer value.
Example: Prior to computer loss, 3DM Core Power And Flow Log power = 1995, manual heat balance power = 2024, and no system changes have taken place. Since a difference is known to exist in the manual heat balance due to data inaccuracies and no system changes have occurred, then 3DM power is the same as heat balance power and the baseline difference is set equal to 29 MWth. Future heat balances that indicate greater than this 29 MWth difference would require power reductions equal to the amount that is greater than the baseline difference.

C. Incorrect: A power reduction would be required if calculated power exceeded 2040 MWt. Plausible in that PNPS 2.1.14 requires operator action be taken if instantaneous power exceeds 2055 MWt.

D. Incorrect: Plausible in that the AMAG computer is what allows the plant to reach 2028. If the plant was being maneuvered, power could not exceed 1998. (ARP 905R-F8) Note that the AMAG computer has also been lost.

Technical Reference(s): PNPS 2.1.10 COMPUTER DATA AND ALARMS (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.3	
	Importance Rating	3.7	

Conduct of Operations: Knowledge of shift or short-term relief turnover practices.

Proposed Question: RO # 67

Shift turnover is in progress. While offgoing personnel are conducting the required control room panel walkdowns with their reliefs, who is normally responsible for maintaining parameter control and control room oversight IAW PNPS 1.3.34, Operations Administration Policies and Processes?

A. Parameter Control – Offgoing BOP
Control room oversight – Offgoing 3RD SRO

B. Parameter Control – Offgoing BOP
Control room oversight – Offgoing Shift Manager

C. Parameter Control – Offgoing 905 Panel Operator / ATC
Control room oversight – Off going 3RD SRO

D. Parameter Control – Offgoing 905 Panel Operator / ATC
Control room oversight – Offgoing Shift Manager

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The BOP leads the oncoming ROs in the control room walkdown.

B. Incorrect: The BOP leads the oncoming ROs in the control room walkdown. The off-going SRO relieves the off-going CRS and assumes control room oversight.

C. Correct: Per 1.3.34, the offgoing Third SRO should normally relieve the offgoing CRS and assume responsibility for oversight of the Control Room. The offgoing C905 Operator will normally maintain parameter control during the Control Room panel walkdown.

D. Incorrect: The off-going SRO relieves the off-going CRS and assumes control room oversight.

Technical Reference(s): PNPS 1.3.34 Section 6.7.3.5 (Attach if not previously provided)
Control Room Panel Walkdown

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:

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

10 CFR Part 55 Content: 55.41 10
55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.44	
	Importance Rating	4.2	

Equipment Control: Ability to interpret control room indications to verify the status and operation of a system, and understand how operator actions and directives affect plant and system conditions.

Proposed Question: RO # 68

During ATWS conditions the following conditions exist:

- The Standby Liquid Control (SLC) Switch is in the SYS "A" position
- The Standby Liquid pump discharge pressures is 1400 psig
- The "A" amber Squib Valve Continuity Light, 1106A, is LIT.

In accordance with PNPS 2.2.24, Standby Liquid Control System, and with these indications, SLC__ (1) __ injecting into the RPV and __ (2) __.

A. (1) IS
(2) the SLC control switch shall then be placed in the "B" position to obtain SLC flow to the RPV at rated capacity.

B. (1) IS NOT
(2) the SLC control switch shall then be placed in the "B" position which should result in SLC flow to the RPV at the rated capacity.

C. (1) IS
(2) No further action is required to obtain full SLC flow to the RPV at rated Capacity.

D. (1) IS NOT
(2) the SLC control switch shall then be placed in the "B" position which should result in SLC flow to the RPV but only at half the rated capacity.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – SLC is not injecting. The switch is placed in "B" if it has been determined that SLC is not injecting

B. Correct – With the Squib Valve Continuity Light, 1106A, LIT, the valve has not fired and there is no flow. Also, discharge pressure should be slightly higher than reactor pressure (not 1400 psig). By procedure, the control switch is moved to the "B" position and SLC should inject at rated.

C. Incorrect – SLC is not injecting

D. Incorrect – The systems are redundant 100% capacity systems

Technical Reference(s): PNPS 2.2.24 Section 7.2 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41 6

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	2	
	K/A #	2.2.12	
	Importance Rating	3.7	

Equipment Control: Knowledge of surveillance procedures

Proposed Question: RO # 69

Given the following:

- Turbine testing is being conducted IAW PNPS 8.A.9-1, "Turbine Test Weekly";
- Exercising of the Emergency Governor from the control room is being performed;
- The EMER TRIP SYS TEST C/S on Panel C2 has just been moved to the RESET position after previously tripping the Emergency Trip System (ETS).

Prior to pushing down on the EMER TRIP SYS TEST C/S and completing the exercising, a Caution in the surveillance requires you to re-verify that:

- The red EMER TRIP RESET light remains ON.
- The green ETS TRIPPED light remains OFF.

Which one of the following describes the reason for this Caution?

If the operator pushes down on the EMER TRIP SYS TEST C/S and the appropriate indications were not present then _____.

A. the turbine will immediately overspeed and trip.

B. the turbine will immediately trip because the Emergency Trip Oil (ETO) header becomes depressurized

C. the turbine overspeed trip function will remain disabled, although no annunciation warns the operator of this condition. The turbine will not trip

D. ONLY the mechanical turbine overspeed function will still operate normally because the controls at the Front Standard are in their normal positions.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – the turbine will trip but not overspeed

B. Correct – IAW MHC Reference Text - Not clearing the trip condition prior to restoring the switch to its normal position will immediately drain the ETO, causing the Steam Admission valves to close and the turbine to trip immediately.

C. Incorrect –there is an associated annunciator (C2L-A2- "OVERSPEED TRIP") and it should be clear

D. Incorrect – IF the Annunciator is not clear the turbine will trip regardless of the overspeed trip function availability

Technical Reference(s): MHC reference Text pages 10 & 11 (Attach if not previously provided)
PNPS 8.A.9-1 – section 8.1

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Pilgrim NRC 2002
Modified Bank # (Note changes or attach parent)
New

Question History: Last NRC Exam: 2002

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

10 CFR Part 55 Content: 55.41 7
55.43

Comments

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	1	
	K/A #	2.1.32	
	Importance Rating	3.8	

Ability to explain and apply system limits and precautions.

Proposed Question: RO # 70

Given the following:

- RCIC is injecting and maintaining adequate core cooling;
- Current RCIC operating parameters are as follows:
    - RCIC Controller position: Auto
    - RCIC injection flow: 200 gpm
    - RCIC Turbine speed: 1500 RPM

IAW PNPS 2.2.22.5, RCIC INJECTION AND PRESSURE CONTROL, which one of the following actions is required and why?

A. Increase injection flow. Flow is too low for stable flow indication.

B. Shift controller to manual. Flow is too low for stable, automatic flow control.

C. Increase turbine speed by increasing injection flow. There is inadequate turbine oil pressure at this RPM.

D. Change turbine speed by increasing or decreasing injection flow. The turbine is operating at a critical speed causing excessive vibration.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Per PNPS 2.2.22.5 precaution [5], oscillations in flow indication do not occur till 100 gpm.

B. Correct: Per PNPS 2.2.22.5, precaution [2], if flow rate is to be less than 225 GPM, the flow controller should be in manual mode due to oscillations of the flow controller at low system flows. Page 15 of the procedure directs that if flow is to be operated less than 225 GPM, the controller is to be placed in manual.

C. Incorrect: Per PNPS 2.2.22.5, precaution [4], adequate oil pressure is ensured down to a speed of 1000 RPM.

D. Incorrect: There is no precaution regarding critical speeds of the turbine. There is a precaution regarding operation less than 2000 RPM involving the potential for water hammer in the exhaust line. However this operation is allowed if required for adequate core cooling and speed is maintained above 1000 RPM.

Technical Reference(s): PNPS 2.2.22.5, RCIC INJECTION AND PRESSURE CONTROL, Precautions and also page 15. (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 8, 10

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.11	
	Importance Rating	3.8	

Radiation Control: Ability to control radiation releases.

Proposed Question: RO # 71

Which one of the following describes the EOP-05, Radioactivity Release Control, actions taken by the operators to mitigate the consequences of an unmonitored release?

A. Initiate CRHEAFs
Start all available Turbine Building Roof Exhaust Fans

B. Start all available Turbine Building Roof Exhaust Fans
Secure the Turbine Basement Exhaust Fans

C. Start all available Turbine Building Roof Exhaust Fans
Start all available Turbine Basement Exhaust Fans

D. Place Control Room Ventilation in recirculation mode.
Secure the Turbine Building Roof Exhaust Fans

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW EOP-05 steps RR2 and RR3

B. Incorrect – securing TB basement exhaust fans is not referenced in the EOP and would not mitigate an unmonitored release

C. Incorrect –These fans should already be in service and this is not an action listed in EOP-05

D. Incorrect – Placing CR ventilation in recirculation mode is permitted to prevent outside air from entering which may contain fumes and/or smoke but this is not appropriate during a radioactive release. The roof exhausters are started, not secured.

Technical Reference(s): EOP-05 steps RR2 and RR3 (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 7

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.1	
	Importance Rating	4.6	

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

Proposed Question: RO # 72

The plant has scrammed from rated power with the following conditions:

- Drywell pressure is 2.4 psig and rising due to a small coolant leak
- Bulk Drywell temperature is 162°F and rising slowly
- RPV level dropped to +8 inches and is now rising
- Secondary Containment ΔP is at +0.6 inches water
- All control rods are at position 00 EXCEPT control rod 26-35 which is at position 48

Which of the following EOP entries are required?

(1) EOP-01, RPV Control
(2) EOP-02, RPV Control - Failure To Scram
(3) EOP-03, Primary Containment Control
(4) EOP-04, Secondary Containment Control

A. (1), (2) and (3) ONLY

B. (1), (3) and (4) ONLY

C. (2), (3) and (4) ONLY

D. (1), (2), (3) and (4)

Proposed Answer: B

Explanation (Optional):

A. Incorrect – EOP-02 is not entered as the reactor will remain shutdown under all conditions with just one rod out. EOP-04 is also entered

B. Correct: EOP-01 is entered due to RPV Level and drywell pressure, EOP-03 is entered on High drywell pressure and temperature, EOP-04 is entered due to high Secondary Containment ΔP

C. Incorrect – EOP-01 is also entered. Plausible in that if EOP-02 is entered, EOP-01 is exited.

D. Incorrect: EOP-02 is not entered as the reactor will remain shutdown under all conditions with just one rod out.

Technical Reference(s): EOPs 01,02,03 and 04 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS 9625

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.29	
	Importance Rating	3.1	

Emergency Procedures/Plan: Knowledge of the emergency plan.

Proposed Question: RO # 73

An Unusual Event has been declared following a complete loss of off-site power during a winter snow storm.

In accordance with EP-IP-100, Emergency Classification and Notification, which one of the following describes the maximum time limitation for notifications?

The Commonwealth and Local Authorities must be notified ____(1)____ and the NRC must be notified ____(2)____.

A. (1) within 15 minutes after event declaration.
(2) no later than 1 hour after event declaration.

B. (1) no later than 1 hour after event declaration.
(2) within 15 minutes after event declaration.

C. (1) within 15 minutes after event declaration.
(2) within 15 minutes after event declaration.

D. (1) no later than 1 hour after event declaration.
(2) no later than 1 hour after event declaration.

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW EP-IP-100 Att.4 sheets 5 and 7.

B. Incorrect – NRC notification is no later than 1 hour after event declaration. Commonwealth is within 15 minutes.

C. Incorrect – NRC notification is no later than 1 hour after event declaration.

D. Incorrect – The commonwealth must be notified within 15 minutes.

Technical Reference(s): EP-IP-100, Att. 4 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: EP-IP-100 Att.4 sheets 5 and 7 (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	3	
	K/A #	2.3.12	
	Importance Rating	3.2	

Radiological Control: Knowledge of radiological safety procedures pertaining to licensed operator duties, such as containment entry requirements, fuel handling responsibilities, access to locked high-radiation areas, aligning filters, etc.

Proposed Question: RO # 74

An operator must enter an area with a dose rate of 1200 MR/hr to perform a task.

In accordance with EN-RP-101, Access Control For RCAs, which one of the following describes the MINIMUM monitoring and radiological controls when accessing the area?

A DLR / TLD, an Electronic Dosimeter, approved RWP and…

(1) Continuous guarding of the entrance to prevent unauthorized entry
(2) Radiation Protection Supervision OR Lead Technician approval
(3) Continuous RP coverage

A. (2)

B. (1) and (3)

C. (2) and (3)

D. (1), (2) and (3)

Proposed Answer: D

Explanation (Optional):

A. Incorrect – A continuous door guard and RP coverage is also required.

B. Incorrect: Radiation Protection Supervision OR Lead Technician approval is also required

C. Incorrect – A continuous door guard is also required.

D. Correct - An area that has a dose rate of 1200 mr/hr is classified as a Locked High Rad Area (LHRA). Per EN-RP-101, Section 5.5, in order to access a LHRA, each person entering a Locked High Radiation Area SHALL have a DLR, an alarming direct reading dosimeter (Electronic Dosimeter), approved RWP, RP Lead technician or RPS approval

and continuous RP coverage. This procedure also specifies that while LHRAs are open, the access to the LHRA SHALL be controlled in accordance with site-specific Technical Specifications. PNPS Tech Specs specify that LHRA areas shall be locked or continuously guarded to prevent unauthorized entry.

Technical Reference(s): EN-RP-101, Section 5.5 (Attach if not previously provided)

Tech Specs Administrative Controls 5.7.2,

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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 12

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #	4	
	K/A #	2.4.49	
	Importance Rating	4.6	

Emergency Procedures/Plan: Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.

Proposed Question: RO # 75

Given the following conditions:

- A loss of feedwater heating results in minor fuel damage
- The 13 minute Off-Gas timer has started but has NOT timed out
- Annunciator "RECOMBINER TEMP HI/LO" CP-600L-A4 alarms
- Recombiner Temperature is 1020°F and rising

Which one of the following actions is required by plant procedures?

A. Manually scram the reactor and enter PNPS 2.1.6, Reactor Scram

B. Commence a normal plant shutdown IAW PNPS 2.1.5, Controlled Shutdown From Power

C. Lower power using Reactor Recirc Pumps and Reverse Order of the Pull Sheet (ROPS) rods until Recombiner Temperature lowers below the alarm setpoint IAW PNPs 2.4.55, Augmented Offgas Explosion.

D. Lower power using Reactor Recirc Pumps and Rapid Power Reduction Rods (RPR) rods until Recombiner Temperature lowers below the alarm setpoint IAW PNPS 2.4.141, Abnormal Recombiner Operation.

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW PNPS 2.4.141 immediate action step

B. Incorrect – a scram is required

C. Incorrect – although this action may lower recombiner temperature, a scram is required IAW 2.4.141

D. Incorrect – although this action may lower recombiner temperature, a scram is required IAW 2.4.141

Technical Reference(s): PNPS 2.4.141, Sect 3.0 [1] pg 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Pilgrim NRC 2002

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2002

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41 10

55.43

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295031 EA2.01	
	Importance Rating		4.6

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

Proposed Question: SRO Question # 76

EOP-01 and EOP-04 execution is in progress due to an un-isolable steam leak on the 51' of the reactor building. The following conditions exist:

- Reactor pressure is being maintained in a band of 450-550 psig
- Reactor level as indicated on Narrow Range Level Indicators LI-263-100A and B is -42 inches and slowly rising
- Reactor level as indicated on Fuel Zone Level Indicators LI-263-106A and B is -155 inches and slowly lowering
- RCIC is the only high pressure systems available and is injecting
- Torus water temperature is 100 degrees and rising
- RHR Pumps A & B are running on minimum flow.
- Torus cooling valves isolated 1 minute ago.
- Both Recirc Pumps are tripped
- The reactor building cannot be accessed due to high temperature conditions throughout.

Given the above:

ACTUAL RPV Level _____ (1) _____ and the required action is _____ (2) _____?

	(1) **Actual Level**	(2) **Required Action**
A.	is -42 inches	Place both RPV LEVEL OVERRIDE keylock switches in override and re-establish torus cooling. Restore RPV level to +12 and +45 inches using RCIC.
B.	is -125 inches	Align all available low pressure ECCS for injection with their pumps running and if level continues to lower, enter EOP-17, Emergency RPV Depressurization.
C.	cannot be determined	Immediately enter EOP-17, Emergency RPV Depressurization and restore level using RHR "A" and

"B".

D. cannot be determined — Exit EOP-01 and enter EOP-16, RPV Flooding and commence flooding the RPV using all available low pressure systems.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Per EOP Caution 1, the minimum useable level for the Narrow Range Instruments is -35 inches. Due to the high Rx Building temperatures (Rx Building not accessible) it must be assumed that level is below the variable leg tap and that the instrument is showing a rising level due to reference leg heatup. The fuel zone instrument is not susceptible to this issue and is reflecting actual level changes. If actual level was -42 inches and the fuel zones were deemed not reliable, this would be the action.

B. Correct: The fuel zone instruments are still reliable. Even though there are high reactor building temperatures there is no indication of flashing. However, they are still off calibrated conditions. Using PNPS 2.2.80, Attachment 8, Figure 2, and given a pressure of 450 - 550 psig, TAF is -155 inches. Therefore actual level is at TAF or, -125 inches. Per EOP-01 the direction is to line up systems for emergency depressurization. EOP-17 must be entered when level cannot be restored and maintained above -150". Per PNPS 5.3.35.2 the direction is to Enter EOP-17 as soon as level drops below TAF and there is reasonable assurance that the low pressure systems can recover level.

C. Incorrect: Actual water level can be determined and is -125 inches.

D. Incorrect: Actual water level can be determined and is -125 inches.

Technical Reference(s): PNPS 2.2.80, Attachment 8, Figure 2 (Attach if not previously provided)

EOP-01, Level leg and EOP Caution 1.

Proposed References to be provided to applicants during examination: Figure 2 of Attachment 8 of PNPS 2.2.80

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

	New	X
Question History:		Last NRC Exam:
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	5
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295004 AA2.03	
	Importance Rating		2.9

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER: Battery voltage

Proposed Question: SRO Question # 77

Given the following:

- The plant is at rated conditions
- 250 VDC Backup Charger D15 is out of service
- "A" EDG is out of service for scheduled maintenance. The plant is on day 2 of a 14 day LCO in accordance with Tech Spec 3.5.F.1

Then..

- The supply breaker to 250 VDC Normal Charger D13 trips
- Alarm 250V DC UNDERVOLTAGE, C3RC-A6, annunciates
- 250 VDC battery voltage is reported as 208 VDC
- Action is immediately initiated to repair the Normal Charger supply breaker AND HPCI is manually isolated.

Assuming conditions do not improve how long can the plant continue to operate?

A. 24 hours

B. 3 days

C. 7 days

D. 12 days

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Plausible in that this would be correct if TS 3.5.F was not met. Although HPCI is inoperable and is a Core Cooling System it is not a low pressure system. 3.5.F states that during any period when one emergency diesel generator (EDG) is inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless such EDG is sooner made operable, provided that all of the **low pressure** core and containment cooling systems shall be operable, and the remaining EDG shall be operable in accordance with 4.5.F.1. If this requirement cannot be met, an orderly shutdown shall be initiated and the reactor shall be placed in the Cold Shutdown Condition within 24 hours.

B. Correct: The 250 VDC battery is inoperable. Per Tech Spec bases, battery voltage must be greater than 210 VDC for the battery to be operable. Tech Spec 3.9.B.5 specifies that from and after the date that the 250 volt battery system is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding three days 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. 3.5.F remains satisfied.

C. Incorrect: The 250 VDC battery is inoperable. Tech Spec 3.9.B.5 allows continued operation for only three more days. Plausible if the candidate thinks that a loss of the 250 VDC system affects the RHR system which is a 7 day LCOs. Although 250 VDC is the power supply to RHR valve MO-1001-47, a loss of power does to this valve does not render RHR inoperable.

D. Incorrect: The 250 VDC battery is inoperable. Plausible in that if the battery was considered operable, the EDG LCO would be limiting.

Technical Reference(s): Tech Spec 3.9.B and associated bases. (Attach if not previously provided)
Tech Spec 3.5.F

Proposed References to be provided to applicants during examination: 3.9.B –No bases
3.5.F – No Bases.

Learning Objective: (As available)

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

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

	Comprehension or Analysis		X
10 CFR Part 55 Content:	55.41		
	55.43	2	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295003 AA2.04	
	Importance Rating		3.7

Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER: System lineups

Proposed Question: SRO Question # 78

Given the following:

- A large break LOCA has occurred
- Electric plant status is as follows:
- All offsite power has been lost
- 4160 VAC Bus A5 has locked out
- 4160 VAC Bus A6 is energized via the "B" EDG
- RHR Pumps B & D are injecting at full capacity
- Core Spray Pump B is injecting at full capacity
- RPV Level is being maintained at -100 inches
- Torus water temperature is at 135°F and rising slowly.

Which one of the following RHR system lineups is required to mitigate the rising torus temperature?

A. In accordance with PNPS 2.2.19.5, RHR Modes of Operation for Transients, close RBCCW nonessential block valves to maximize RBCCW flow to the RHR heat exchanger.

B. In accordance with PNPS 2.4.A.5, Loss Electrical Bus A5, secure RHR injection and place RHR loop B in 2 pump torus cooling mode to maximize heat rejection to RHR heat exchanger

C. In accordance with PNPS 2.4.A.5, Loss Electrical Bus A5, secure RHR injection and place RHR loop B in single pump torus cooling mode to maximize heat rejection to RHR heat exchanger.

D. In accordance with PNPS 2.2.19.5, RHR Modes of Operation for Transients, close the heat exchanger bypass valve on RHR loop B and maintain both RHR Pumps running to maximize heat transfer to the RHR heat exchanger.

Proposed Answer: A

Explanation (Optional):

A. Correct – 2.2.19.5 and EOP-03 direct that if only one loop of RHR and/or RBCCW is available, then RBCCW flow is to be maximized for the available loop when torus temperature exceeds 130 degrees and a "major" LOCA is in progress. This is done by isolating the non-essential loads.

B. Incorrect – RHR injection is required to maintain level

C. Incorrect – RHR injection is required to maintain level

D. Incorrect – The RHR H/X only has the capacity for one pump.

Technical Reference(s): PNPS 2.2.19.5, RHR Modes of Operation for Transients, pages 18 (Attach if not previously provided) and 19

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # TADS 6579

Modified Bank #

New

Question History: Last NRC Exam: Question #79, 2009

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	600000 2.4.41	
	Importance Rating		4.6

Emergency Procedures / Plan: Knowledge of the emergency action level thresholds and classifications. (Plant Fire On-site)

Proposed Question: SRO Question # 79

Given the following:

- The plant is shutdown and cooling down for a refuel outage
- Reactor pressure is 500 psig and lowering.

The following sequence then occurs:

Time 00:00:	• Startup transformer locks out due to an electrical fault • "A" Diesel Generator starts and re-energizes A5 • "B" Diesel Generator fails to start and the Shutdown Transformer Re-energizes A6
Time 00:05:	• The Shift Manger reviews EALs
Time 00:30:	• Fire alarms received for upper switchgear room • Bus A5 locks out • Fire Brigade Chief reports fire in bus A5 and that the Brigade is actively fighting the fire. The Chief requests Plymouth Fire Fighting assistance.
Time 00:35:	• The Shift Manager calls Plymouth Fire and reviews EALs
Time 00:45:	• Fire Brigade Chief reports that Plymouth Fire is on scene and fighting the fire.
Time 00:50:	• The Shift Manager reviews EALs
Time 00:60:	• Fire extinguished. Chief reports fire was limited to A5 which is damaged extensively. • The Shift Manager reviews EALS

The Shift manager was required to declare …

A. An Unusual Event at time 00:05
An Alert at time 00: 50

B. An Unusual Event at time 00:35
A Site Area Emergency at time 00: 60

C. An Unusual Event at time 00:35
No other declarations were required

D. An Alert at time 00: 50
A Site Area Emergency at time 00: 60

Proposed Answer: C

Explanation (Optional):

A. Incorrect: An EAL has not yet been exceeded at time 00:05. Plausible in that a UE is required if all off-site power is lost. However the Shutdown Transformer is still an available off-site power source negating the necessity to declare the UE. An Alert is also not required. Plausible in that an Alert would be required if the fire was burning out of control. There was no evidence that the fire was "out of control" and spreading.

B. Incorrect: A UE was exceeded when off-site fire fighting assistance was requested per EAL 7.2.1.1. However a SAE was never exceeded during the event. Plausible in that "any fire which has affected the ability of two or more safety systems (Table 7-1) to perform their intended function and poses a significant potential for release of radioactivity" would result in a SAE per EAL 7.2.1.3. Although multiple safety systems are affected (Core Spray, RHR, SBGT etc.) the plant can still achieve cold shutdown and there is no significant potential for a release.

C. Correct: The SM was required to declare UE when off-site fire fighting assistance was requested per EAL 7.2.1.1 at time 00:35. No other EALs were exceeded.

D. Incorrect: The SM was required to declare UE when off-site fire fighting assistance was requested per EAL 7.2.1.1 at time 00:35. An Alert was also not required as discussed above. A SAE was also never exceeded as discussed above.

Technical Reference(s): EP-IP-100.1 EMERGENCY ACTION LEVELS (EALs), Attachment 1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: EP-IP-100.1 EMERGENCY ACTION LEVELS (EALs), Attachment 1

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295026 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. (Suppression Pool High Water Temp)

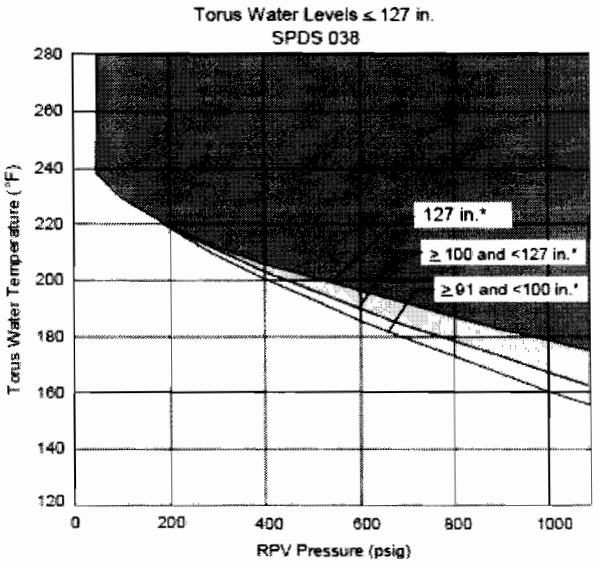
Proposed Question: SRO Question # 80

During severe accident conditions the following conditions exist:

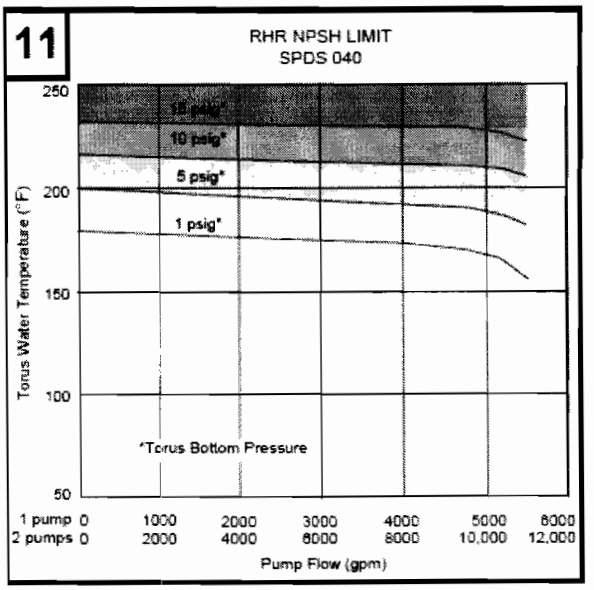
- Drywell Spray has just been initiated using "A" and "B" RHR pumps
- "C" and "D" RHR pumps are secured
- RHR flow in each loop is 5000 gpm.
- RPV Pressure: 700 psig
- Torus Bottom Pressure: 18 psig and lowering
- Torus Water Level: 110 inches and steady
- Torus Water Temperature: 190 degrees and steady

What is the status of the Heat Capacity Temperature Limit (HCTL) and the RHR NPSH limitations based on the above parameters?

Heat Capacity Temperature Limit



RHR NPSH Limit



A. The HCTL IS being exceeded. The RHR NPSH will be exceeded when Torus bottom pressure lowers to below 5 psig.

B. The HCTL IS being exceeded. The RHR NPSH will NOT be exceeded provided there is no further degradation in Torus Water Temperature or Torus Water Level.

C. The HCTL Is NOT being exceeded. The RHR NPSH will be exceeded when Torus bottom pressure lowers to below 5 psig.

D. The HCTL Is NOT being exceeded. The RHR NPSH will NOT be exceeded provided there is no further degradation in Torus Water Temperature or Torus Water Level.

Proposed Answer: A

Explanation (Optional):

A. Correct: At an RPV Pressure of 700 PSIG and a Torus level of 110 inches the HCTL is exceeded when torus water temp exceeds 185 degrees. The NPSH for each pump per loop running at 5000 gpm and 190 degrees torus water temperature will be exceeded when torus bottom pressure lowers to less than 5 psig. Given a torus water level of 110 inches, torus bottom pressure will lower as low as ~ 4 psig following drywell spray initiation.

B. Incorrect: The RHR pump NPSH limit will eventually be exceeded as drywell sprays lower the pump over pressure. Plausible in that if the "2 pump value" of 5000 gpm is used it would not appear that the limit would be exceeded as the torus water temperature limit would be ~ 195 degrees. (drywell sprays could not lower the torus bottom pressure to any lower than ~ 4 psig due to the weight of the water).

C. Incorrect: At an RPV Pressure of 700 PSIG and a Torus level of 110 inches the HCTL is exceeded when torus water temp exceeds 185 degrees. Plausible in that if the torus water level line of 127 inches is used, the HCTL would not be exceeded until ~ 195 degrees.

D. Incorrect: At an RPV Pressure of 700 PSIG and a Torus level of 110 inches the HCTL is exceeded when torus water temp exceeds 1185 degrees. Plausible in that if the torus water level line of 127 inches is used, the HCTL would not be exceeded until ~ 195 degrees.

Technical Reference(s): EOP-11 Figures, Cautions and Icons (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:

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

10 CFR Part 55 Content:	55.41	
	55.43	5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	295019 2.4.4	
	Importance Rating		4.7

Emergency Procedures / Plan: Ability to recognize abnormal indications for system operating parameters which are entry-level conditions for emergency and abnormal operating procedures. (Partial or Total Loss of Inst. Air)

Proposed Question: SRO Question # 81

Given the following:

- The reactor is at 100% power
- Annunciator C904LC-F3, AIR/N2 TO DRYWELL TROUBLE, alarms
- A Control Room operator reports that PI-4348, NITROGEN SUPPLY DRYWELL EQUIP SUPPLY PRESSURE, on Panel C7 reads 60 psig and continuing to lower
- Drywell pressure is 2.0 psig and continuing to rise.

Which one of the following is required?

A. Direct that instrument air be aligned to supply drywell pneumatics and isolate the nitrogen supply.

B. Direct that the reactor be scrammed in accordance with PNPS 2.1.6, Reactor Scram and isolate drywell pneumatics.

C. Direct that the torus be vented to maintain pressure less than 2.2 psig and align instrument air to augment the nitrogen supply to the drywell pneumatics.

D. Direct that the drywell be vented to maintain pressure less than 2.2 psig and initiate a shutdown in accordance with PNPS 2.1.5, Controlled Shutdown From Power.

Proposed Answer: B

Explanation (Optional):

A. Incorrect – There is indication of a leak and aligning instrument air will just continue to pressurize the containment.

B. Correct. Symptoms indicate a drywell pneumatic supply line break in the drywell. 2.4.21 requires a reactor scram when pressure approaches 2.2.

C. Incorrect – Incorrect – there is no direction to vent the containment.

D. Incorrect – there is no direction to vent the containment. Loss of pneumatics will not support a controlled shutdown from power (MSIVs will close)

Technical Reference(s): PNPS 2.4.21, DOUBLE ENDED BREAK OF THE 3-INCH INSTRUMENT PNEUMATIC LINE IN THE DRYWELL, page 3 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS # 3674

Modified Bank # (Note changes or attach parent)

New

Question History: 2009 Audit Exam, question 90 Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	1	
	K/A #	700000 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. (Generator Voltage and Electric Grid Disturbances)

Proposed Question: SRO Question # 82

Given the following:

- The plant is at 100% power
- The Shutdown Transformer is out of service
- A grid disturbance results in momentary LINE 342 and 355 UNDERVOLTAGE alarms, C3R-A7 and C3R-A8
- The Main Generator Voltage Regulator trips to manual
- Main Generator voltage and MVAR loading is stabilized using the Manual Voltage Regulator
- ISO New England/NSTAR notifies PNPS that voltage cannot be maintained ≥ 343.5 KV if Pilgrim were to trip.

Which one of the following is correct regarding offsite notifications IAW PNPS 2.4.144, Degraded Voltage?

Notify:

A. ISO New England within 30 minutes that the Main Generator Voltage Regulator is in Manual. NRC notification is not required.

B. ISO New England within 60 minutes that the Main Generator Voltage Regulator is in Manual. NRC notification is not required.

C. ISO New England within 60 minutes that the Main Generator Voltage Regulator is in Manual. NRC notification of plant status is required.

D. ISO New England within 30 minutes that the Main Generator Voltage Regulator is in Manual. NRC notification of plant status is required.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: NRC notification is required IAW 10CFR50.72(b)(3)(v),

B. Incorrect: IAW the immediate action of PNPS 2.4.144, PNPS is to notify the ISO within 30 minutes that the regulator is in manual – not 60 minutes. Additionally, NRC notification is required IAW 10CFR50.72(b)(3)(v).

C. Incorrect: IAW the immediate action of PNPS 2.4.144, PNPS is to notify the ISO within 30 minutes that the regulator is in manual – not 60 minutes.

D. Correct: IAW the immediate action of PNPS 2.4.144, PNPS is to notify the ISO within 30 minutes that the regulator is in manual. The Startup Transformer is also required to be declared inoperable when PNPS is notified that proper voltage cannot be maintained post trip. Therefore both the Shutdown Transformer and the Startup Transformer are now inoperable. The note on page 14 of PNPS 13.12 directs that a 10CFR50.72(b)(3)(v), should be made in this situation. This is also specified in the subsequent actions of PNPS 2.4.144.

Technical Reference(s): PNPS 1.3.12, Attachment 12, sheets 2 through 10 (Attach if not previously provided)

PNPS 2.4.144, Immediate Action

Tech Spec 3.9.B.2

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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 1 , 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295032 EA2.02	
	Importance Rating		3.5

Ability to determine and/or interpret the following as they apply to HIGH SECONDARY CONTAINMENT AREA TEMPERATURE: Equipment operability

K/A Justification: Determination of equipment operability is an SRO function at PNPS.

Proposed Question: SRO Question # 83

Given the following conditions:

- A HPCI steam leak has occurred in the reactor building.
- Efforts to isolate the leak are unsuccessful.
- HPCI Turbine area is 182 degrees F.
- HPCI Piping area-Torus Compartment is 275 degrees F.

Emergency Depressurization:

EOP-04, Table L MAXIMUM SAFE OPERATING VALUES – Selected Portion
(Temperature areas are separated by dashed lines)

Temperature °F		
RCIC Piping Area -Torus Compt	TS-1340-8A	258
RCIC Turbine Area -Stairwell	TS-1340-8B	175
HPCI Piping Area -Torus Compt	TS-2340-8A	258
HPCI Turbine Area -17 ft El.	TS-2340-8B	175
RWCU Piping Area -36 ft El. Mezzanine	TS-1290-26H	238
RCIC Tip Room -23 ft El.	TS-1340-8C	224
Main Steam Tunnel -23 ft El.	TS-260-18A	289
HPCI Piping Area -23 ft El. ("B" RHR Valve Room)	TS-2340-8C	309
RHR "B" & "D" Pump Area -Stairwell	TS-1040-16A	200
RHR "A" & "C" Pump Area -6 ft El.	TS-1040-16B	200

A. is currently required because the integrity of the secondary containment is threatened.

B. is currently required because the continued operability of safety related equipment is threatened.

C. will be required if the water level in the HPCI compartment rises to 8 inches because the integrity of the secondary containment is threatened.

D. will be required if HPCI Piping Area-23 ft Elevation (B RHR Valve Room) exceeds 309 degrees F because the continued operability of safety related equipment is threatened.

Proposed Answer: D

Explanation (Optional):

A. Incorrect: Emergency depressurization (ED) is not currently required. Plausible in that both temperatures are above their Max Safe Operating Values but they are within the same area. ED is performed if Max Safe values are exceeded in 2 different areas.

B. Incorrect: Emergency depressurization (ED) is not currently required. Plausible in that both temperatures are above their Max Safe Operating Values but they are within the same area. ED is performed if Max Safe values are exceeded in 2 different areas.

C. Incorrect: ED is required when the same parameter exceeds the Max Safe Values in 2 or more areas. Plausible in that the 8 inches of water is above the Max Safe value but a different parameter.

D. Correct: When the B RHR valve room temperature exceeds 309 degrees, Max Safe Values are now exceeded in two different areas and ED is required. The ED is required to maintain secondary containment integrity but also because the operability of safety related equipment is threatened.

Technical Reference(s): EOP-04, Secondary Containment Control. (Attach if not previously provided)
BWR OG EPGs Appendix B page B-8-14.

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # Western Tech Bank 2104

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	
	55.43	1, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	500000 2.4.6	
	Importance Rating		4.7

Emergency Procedures / Plan: Knowledge of EOP mitigation strategies. (High CTMT Hydrogen Conc)

Proposed Question: SRO Question # 84

Given the following:

- A large LOCA inside the drywell occurs. EOP-01 and EOP-03 are entered
- The Hydrogen and Oxygen analyzers are placed in service.
- One hour later, an increase in hydrogen concentration is detected on both analyzers.

At what point are EOP-01 and EOP-03 exited and Primary Containment Flooding commenced?

Not until Hydrogen concentration cannot be maintained below.....

A. 1%, regardless of Oxygen concentration

B. 4%, regardless of Oxygen concentration

C. 6%, regardless of Oxygen concentration

D. 6%, AND Oxygen concentration exceeds 5%

Proposed Answer: B

Explanation (Optional):

A. Incorrect: EOP-03 specifies that the EOPs and Containment Flooding (SAGs) be entered when [H2] exceeds 4%. Plausible in that this is the value where containment venting is commenced.

B. Correct: When [H2] is above 4%, step G-5 requires that Primary Containment Flooding be executed. EOP-03 override C-1 directs that EOP-03 be exited and SAGs entered when Primary Containment Flooding is required. EOP-01 override R-1 directs that EOP-01 be exited and SAGs entered when Primary Containment Flooding is required.

C. Incorrect: EOP-03 specifies that the EOPs and Containment Flooding (SAGs) be entered when [H2] exceeds 4%. Plausible in that the is the [H2] associated with a deflagration.

D. Incorrect: SAGS are entered when [H2] exceeds 4%. Plausible in that these are the deflagration limits.

Technical Reference(s): EOP-03, Primary Containment Control (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:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 1, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	1	
	Group #	2	
	K/A #	295015 2.4.18	
	Importance Rating		4.0

Proposed Question: SRO Question # 85

Emergency Procedures / Plan: Knowledge of the specific bases for EOPs. (Incomplete SCRAM)

Given the following:

- The reactor had been operating at 100% power for an extended period when a rising drywell pressure resulted in an automatic scram;
- Several control rods failed to insert;
- EOP-02, Failure to Scram has been entered and the RO is inserting rods manually;
- Reactor Power is on range 2 of the IRMs and lowering;
- RPV Level is being maintained +12 inches to +45 inches utilizing the feed system.
- No boron has been injected;
- An RPV cooldown has commenced utilizing turbine bypass valves.

With these conditions, the reactor operator reports that power has now risen to range 6 of the IRMs and is continuing to slowly rise.

Which one of the following is required and the bases for that action?

A. Terminate injection IAW PNPS 5.3.35.1 ATTACHMENT 35 - STOP AND PREVENT INJECTION CHECKLIST and lower level to below -25 inches in order to decrease the amount of inlet subcooling and thereby reduce power.

B. Stop reducing pressure and stabilize pressure below 1060 psig IAW EOP-02 Step P-5 to allow the negative temperature coefficient and the reactor operator to insert additional rods to shutdown the reactor.

C. Stop reducing pressure and inject Boron IAW PNPS 5.3.35.1 ATTACHMENT 44 - INITIATION OF STANDBY LIQUID CONTROL. Once boron injection has commenced continue reducing pressure because the boron injection can overcome any positive reactivity addition by the cooldown.

D. Terminate injection IAW PNPS 5.3.35., 1 ATTACHMENT 35 - STOP AND PREVENT INJECTION CHECKLIST and lower level as required until power lowers to range 6 of the IRMs in order to reduce the amount of natural circulation in the core. Do not intentionally lower level below -125".

Proposed Answer: B

Explanation (Optional):

A. Incorrect: Injection is not terminated and prevented and level lowered to <-25" unless power is above 3%.

B. Correct: Per EOP-02, step P-7, the direction is to stabilize pressure below 1060 psig when the reactor is not shutdown. PNPS 5.3.35 defines reactor shutdown as being on IRM range 7 or lower and power continuing to lower. Stopping the pressure reduction adds negative reactivity be allowing the reactor to re-pressurize and the operator time to insert more control rods.

C. Incorrect: Once the decision is made to inject boron, further intentional cooldown is not allowed until the cold shutdown boron weight has been injected.

D. Incorrect: Injection is not terminated and prevented and level lowered to control power unless power is above 3% and torus temperature is above the BITT.

Technical Reference(s): EOP-02 step P-7 (Attach if not previously provided)

PNPS 5.3.35, page 13

IG O-RO-03-04-04, page 53 and 54

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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	223002 A2.08	
	Importance Rating		3.1

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: Surveillance testing

Question Justification: In order to answer this question, candidates must integrate system knowledge with their analysis of the conditions provided and then determine that two separate and different action statements apply. Specifically, the candidate must:

1) determine that two of the three instrument trip settings do not meet tech spec requirements;
2) review the reference and determine that a minimum of two instruments is required per trip system;
3) recall the logic arrangement of the trip systems and determine that only two instruments are available and therefore the minimum number required by tech specs is not met (this information is not contained within the reference provided);
4) review the table notes and based on their knowledge of logic arrangement, determine that tripping the trip systems will not cause isolations
5) determine that one trip system must be tripped within one hour for one function but that an exception for the other function must be applied and that the other trip system must be tripped within 12 hours (vice the previous 1 hour).

Based on the above, this question is not a "direct lookup".

Proposed Question: SRO Question # 86

Given the following:

- The plant is at 100% power
- Surveillance procedure 8.M.1-32.1 ANALOG TRIP SYSTEM - TRIP UNIT CALIBRATION - CABINET C2228-A1has just been completed.
- While reviewing the surveillance results the following As-Left Trip Settings are noted:
    - LIS-263-57A Reactor Water Level Low-Low, tripped at – 48 inches
    - LS-263-57A-1 Reactor Water Level Low, tripped at +10.5 inches
    - LS-263-57A-2 Reactor Water Level High, tripped at + 54.2 inches

Based on the above:

(1) Which Channel A-1 PCIS function(s) is(are) inoperable

And

(2) The required action(s) is(are) to:

	Inoperable Function(s)	Required Action(s):
A.	RPV Water Level Low, Low-Low, and High	Trip the Low-Low and High RPV water level A-1 channels within 1 hour Trip the Low RPV water Level A-1 channel within 12 hours
B.	RPV Water Level Low, Low-Low only	Trip the Low-Low RPV water level A-1 channel within 1 hour Trip the Low RPV water Level Channel within 12 hours
C.	RPV Water Level High only	Trip the High RPV water level A-1 channel within 1 hour
D.	RPV Water Level Low, and Low-Low only	Trip the Low and Low-Low RPV water Level Channel within 12 hours

Proposed Answer: B

Explanation (Optional):

A. Incorrect: The RPV High Function is still operable as the Tech Spec requirement is that it be set ≤ 55.4 inches. Plausible in that if it were Inop the required action would be to trip the channel within 1 hour.

B. Correct; The RPV Low level function is required to trip ≥ 11.6 inches. If not, then the required action is to trip the RPV low level channel within 12 hours. The RPV Low-Low level function is required to trip ≥ 46.4 inches. If not then the required action is to trip the RPV low level channel within 1 hour.

C. Incorrect: The RPV High Function is still operable as the Tech Spec requirement is that it be set ≤ 55.4 inches. Plausible in that if it were Inop the required action would be to trip the channel within 1 hour. Additionally the Low and Low-Low functions are inoperable.

D. Incorrect: Although both are inoperable the Low-Low channel must be tripped within 1 hour.

Technical Reference(s): 8.M.1-32.1, section 4.0 step [1] (b), Plant Impact (Attach if not previously provided)
TS Table 3.2.A, Note 1

Proposed References to be provided to applicants during examination: TS Table 3.2.A and

associated notes

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New x

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	215005 A2.02	
	Importance Rating		3.7

Ability to (a) predict the impacts of the following on the AVERAGE POWER RANGE MONITOR/LOCAL POWER RANGE MONITOR SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Upscale or downscale trips

Proposed Question: SRO Question # 87

Given the following:

- The plant is at 100%
- APRM "A" has failed upscale and is bypassed
- Several LPRM detectors that input into APRM channel "C" have been bypassed
- The current status of APRM "C" LPRM inputs are as follows:

o 12-45-A OK	o 36-37-B Bypassed
o 04-21-D OK	o 28-13-C OK
o 28-45-C Bypassed	o 12-29-C OK
o 20-21-B OK	o 44-13-A OK
o 04-37-B OK	o 28-29-A OK
o 36-21-D Bypassed	o 20-05-D OK
o 20-37-D OK	o 44-29-C OK
o 12-13-A OK	o 36-05-B Bypassed

With these initial conditions, LPRM 04-37-B fails downscale.

(1) Prior to any operator action, what will be the impact of the LPRM failure on APRM "C" (assume all LPRMs were reading the same at the time of the failure)

AND

(2) Which of the below actions are required for this condition?

A. (1) APRM "C"'s output will lower.
(2) Enter PNPS 2.4.38, LPRM Failure, bypass LPRM 04-37-B, verify / adjust AGAFs and continue operation without any additional restrictions.

B. (1) APRM "C"'s output will remain the same.
(2) Enter PNPS 2.4.38, LPRM Failure, bypass LPRM 04-37-B, verify / adjust AGAFs and continue operation without any additional restrictions.

C. (1) APRM "C"'s output will lower.
(2) Enter Tech Spec LCO 3.1.1. If APRM "A" or "C" is not restored within the next 12 hours, trip RPS "A" or take other equivalent action authorized by Tech Specs.

D. (1) APRM "C"'s output will remain the same.
(2) Enter Tech Spec LCO 3.1.1. If APRM "A" or "C" is not restored within the next 12 hours, trip RPS "A" or take other equivalent action authorized by Tech Specs.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: Per TS Table 3.1.1, note 13, an APRM is considered operable if it has two LPRMs per level. That is no longer the case in that there is only one remaining "B" level detector. Plausible in that if this is not recognized, PNPS 2.4.38, would have you bypass the LPRM and verify AGAFs since there are still more than 10 LPRMs available (< 10 will generate an INOP trip)

B. Incorrect: The APRM output will lower. Plausible in that unlike the RBM, the APRM does not use downscale trip units which would automatically remove the LPRM from the averaging circuit.

C. Correct: With the additional failure there are too few LPRM inputs for level B (minimum is 2 and only 1 is available) and APRM "C" is inop. With two APRMs INOP associated with RPS "A", Tech Spec Table 3.1.1 condition "a" specifies that RPS "A" must be tripped within 12 hours or actions A or B initiated

D. Incorrect: The APRM output will lower. Plausible in that unlike the RBM, the APRM does not use downscale trip units which would automatically remove the LPRM from the averaging circuit.

Technical Reference(s): TS Table 3.1.1 and associated note 13. (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:	
Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	2, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	203000 2.2.42	
	Importance Rating		4.6

Equipment Control: Ability to recognize system parameters that are entry-level conditions for Technical Specifications. (RHR/LPCI: Injection Mode)

Question Justification: In order to answer this question, candidates must apply previous PNPS system Operating Experience (LER) to the condition presented. This information is not documented in the PNPS Tech Specs or its associated bases but captured in plant procedures that are not provided as references to the candidates. Integrating their knowledge of the operating experience, candidates must recognize that the LPCI function is inoperable and then apply this condition to the existing HPCI LCO. Other distracters are based on a common misunderstanding that the operating experience requires that a Suppression Pool Cooling subsystem of RHR is to be declared inoperable since that is the existing RHR lineup. Therefore this question is not a "direct lookup".

Proposed Question: SRO Question # 88

Given the following:

- The plant is at 100% power
- HPCI has been declared inoperable and is on day 2 of a 14 day LCO
- RHR Loop "A" is placed into torus cooling to support post work testing of the HPCI system

(1) The SRO is required to declare:

AND

(2) The most limiting LCO is:

	Required Declaration	**Most Limiting LCO**
A.	LPCI Inop	7 Day LCO
B.	LPCI Inop	24hr Cold S/D LCO
C.	"A" Suppression Pool Cooling Subsystem Inop	7 Day LCO
D.	"A" Suppression Pool Cooling Subsystem Inop	24hr Cold S/D LCO

Proposed Answer: B.

Explanation (Optional):

A. Incorrect: LPCI is required to be declared INOP. With HPCI already inoperable TS 3.5.C.3 requires that a 24 hour cold S/D LCO be entered.

B. Correct: Whenever the LPCI System is in the Torus Cooling mode of operation LPCI is to be declared inop ((Tech Spec 3.5.A). This is based on 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. With HPCI already inoperable TS 3.5.C.3 requires that a 24 hour cold S/D LCO be entered.

C. Incorrect: LPCI is required to be declared Inop

D. Incorrect: LPCI is required to be declared Inop

Technical Reference(s): TS 3.5.C.3 (Attach if not previously provided)

PNPS 2.2.19, RHR, page 15

Proposed References to be provided to applicants during examination: TS 3.5.C

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 2

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	262002 2.1.20	
	Importance Rating		4.6

Conduct of Operations: Ability to interpret and execute procedure steps. [UPS (AC/DC)]

Proposed Question: SRO Question # 89

Given the following:

- Power has been reduced to 40% with a plant shutdown in progress
- A momentary loss of 120VAC Vital Bus Y-2 occurs when the Vital MG set trips
- Y-2 is now re-energized
- RPV level is 32 inches and slowly rising

Based on these conditions the CRS is required to enter:

A. PNPS 5.3.6, LOSS OF VITAL AC (Y2) and direct resetting the Feed Reg Valves lockups, the Recirc runback, and scoop tube locks.

B. PNPS 5.3.6, LOSS OF VITAL AC (Y2) and direct resetting the Recirc runback and scoop tube locks. Direct Recirc pump speed be increased to stabilize level until the Feed Reg Valves are locked in the condenser compartment.

C. Enter PNPS 2.4.49, FEEDWATER MALFUNCTIONS and direct that the Feed Reg Valves be locked in the Condenser Compartment. Enter PNPS 2.4.36, DECREASING CONDENSER VACUUM and direct re-opening AO-3751, Off Gas Isolation Valve.

D. Enter PNPS 2.4.49, FEEDWATER MALFUNCTIONS and direct that the Feed Reg Valves be locked in the Condenser Compartment. Enter PNPS 2.4.20, REACTOR RECIRCULATION SYSTEM SPEED OR FLOW CONTROL SYSTEM MALFUNCTION and direct resetting the scoop tube locks.

Proposed Answer: A

Explanation (Optional):

A. Correct: PNPS 5.3.6, LOSS OF VITAL AC (Y2) provides direction on how to stabilize the plant after a momentary loss of Y-2. Resetting the FRV lockups will stabilize level. Resetting the runbacks and then the scoop tube locks will restore speed control.

B. Incorrect: RPV level is stabilized by resetting the FRV lockups. Plausible in that raising reactor power would help stabilize the level rise.

C. Incorrect: The Feed Reg Valves can be reset on C905 following a momentary loss of Y-2. Plausible in that PNPS 2.4.49, Feedwater Malfunctions, directs that if the FRVs are locked and the valves are drifting open or closed, that the valves should be locked in the condenser compartment. Procedure 2.4.36 is plausible in that the off-gas isolation valve did go close during the Y-2 loss and would have re-opened once power was restored.

D. Incorrect: PNPS 2.4.49 is plausible for the reason discussed above. PNPS 2.4.20 is also plausible in that the Recirc pumps locked up following the loss of Y-2.

Technical Reference(s): PNPS 5.3.6, Attachment 1 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Q deleted per NRC resolution of facility comment.
T Fish
3/30/11

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	259002 2.1.19	
	Importance Rating		3.8

Ability to use plant computers to evaluate system or component status. (Reactor Water Level Control System)

SRO Level Justification: The Safety Parameter Display System (SPDS) at PNPS is a subset of the overall PNPS process computer also known as EPIC. Whereas all operators utilize the general process computer functions to monitor plant status, the SPDS function is the purview of the SRO. This is stated explicitly in PNPS 2.6.1 EMERGENCY AND PLANT INFORMATION COMPUTER (EPIC) SYSTEM DISPLAY, on page 6, when it states that "The Shift Control Room Engineer (SCRE) is designated as the primary SPDS user". The procedure goes on to state that the primary location for the SPDS displays is to be on one of the Control Room Supervisor (CRS) computer displays. Both the SCRE and CRS positions are only filled by SROs at PNPS.

Although some general features of the EPIC system used by all operators are similar to those within the SPDS there are significant, additional SPDS features that are not used within the process computing functions. For example, EPIC displays alarms and instrument readings using three general colors – green (normal), yellow (approaching an alarm setting), and red (exceeding an alarm setting). SPDS uses these same colors plus, white, cyan, dark blue, and magenta. Additionally for some common colors such as yellow, the color means different things depending on the particular SPDS display. For example, unlike EPIC in general, the displays for critical plant parameters utilize inputs from multiple instruments and go through extensive validation. If the validation algorithms determine that there are insufficient inputs or the inputs are suspect, the SPDS warns the SRO that the display has not been validated and its value is suspect by outlining the value in yellow. A yellow display on EPIC means that the value is approaching an alarm setpoint. In summary, if the SRO candidates based their response to this question on their knowledge of EPIC color conventions they would arrive at an incorrect response.

Proposed Question: SRO Question # 90

Given the following:

- Reactor level is +30" on the feedwater range level indication and steady.
- On the SPDS Critical Plant Variables Display, the digital readout for RPV "ACTUAL LEVEL" is displayed in YELLOW numbers reading +32" and surrounded by a YELLOW border.

The SPDS display means that the SPDS calculated RPV water level .....

A. has exceeded the allowable difference between the calculated value and the output of the feedwater level control instruments.

B. has reached the high level alarm setpoint.

C. has insufficient data to validate the calculation

D. is using at least one bad data input in its calculation.

Proposed Answer: C

Explanation (Optional):

A. Incorrect: The SPDS value for level is calculated via inputs from several instruments. Instruments are compared to one another to determine whether any single instrument has failed but not to the calculated SPDS value.

B. Incorrect: The yellow border means that the calculation has not been validated. Plausible, in that Operational Limit tags will change to yellow when they are approaching an alarm limit. +32 inches is the alarm setting for the feedwater level control system high level alarm.

C. Correct: The yellow border means that the value has not been satisfied. Plausible in that some displays such as SPDS Alarms and parameters plotted on graphs will turn yellow if the measured value is approaching a limit.

D. Incorrect: Bad data would result in a magenta display.

Technical Reference(s): PNPS 2.6.1 EMERGENCY AND PLANT INFORMATION COMPUTER (EPIC) SYSTEM DISPLAYS, page 27. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # TADS 5723

Modified Bank # (Note changes or attach parent)

New

Question History:	Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43 5	

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	214000 A2.02	
	Importance Rating		3.7

Ability to (a) predict the impacts of the following on the ROD POSITION INFORMATION SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Reactor SCRAM

Proposed Question: SRO Question # 91

Given the following:

- The reactor scrammed 2 minutes ago due to low RPV water level.
- RPV water level is currently +5 inches and slowly recovering
- All but six control rods can be verified as having fully inserted.
- The RO reports the following indications for the six rods:
  - Neither the Full In or Full Out lights are illuminated on the full core display.
  - The four rod display for these rods indicate "black-black"
  - EPIC and SPDS Rod Position Displays indicate magenta *** for these six rods
  - CALLRODS indicates "YES"

Which one of the following is correct?

A. Reactor shutdown status has been confirmed. Remain in EOP-01 and direct normal scram recovery actions.

B. Reactor shutdown status cannot be determined. Exit EOP-01 and Enter EOP-02 and attempt to insert the control rods.

C. CALLRODS has determined that the rods did NOT insert. Exit EOP-01 and Enter EOP-02 and attempt to insert the control rods.

D. CALLRODS has been triggered but has not yet confirmed rod insertion. Remain in EOP-01 for an additional minute before making a determination of rod position.

Proposed Answer: A

Explanation (Optional):

A. Correct: These are the RPIS indications for rods that have gone beyond full-in following the scram. The CALL RODS program detected that the six rods have passed through the 04 position (Max Subcritical Bank Withdrawn Position) as indicated by the word "YES". All rods have been inserted beyond the Max Subcritical Bank Withdrawn Position and stabilization efforts are governed by EOP-01

B. Incorrect: These are the RPIS indications for rods that have gone beyond full-in following the scram. The CALL RODS program detected that the six rods have passed through the 04 position (Max Subcritical Bank Withdrawn Position) as indicated by the word "YES". Plausible in that if the rod position could not be confirmed, this would be the required action.

C. Incorrect: The six rods did insert. Plausible if the candidate does not understand the meaning of CALLRODS indications.

D. Incorrect: Call rods has determined that the rods have inserted as described above. Plausible in that Scram procedure 2.1.6 discusses that it may take up to three minutes for the CALL RODS program to determine control rod status.

Technical Reference(s): Scram Procedure 2.1.6, Discussion Section, page 4 (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:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	2	
	K/A #	233000 2.4.9	
	Importance Rating		4.2

Emergency Procedures / Plan: Knowledge of low power / shutdown implications in accident (e.g., loss of coolant accident or loss of residual heat removal) mitigation strategies. (Fuel Pool Cooling/Cleanup)

Proposed Question: SRO Question # 92

Given the following:

- An RFO is in progress; refueling is complete
- The fuel pool is isolated from the reactor basin and reactor basin draindown has commenced.
- The 'B' Fuel Pool Cooling heat exchanger has developed a major tube leak and is isolated.
- Fuel Pool Cooling temperature is 115 degrees F and rising

In accordance with PNPS 2.2.85, FUEL POOL COOLING AND FILTERING SYSTEM, the fuel pool temperature limit will be exceeded when pool temperature reaches and exceeds ____ (1) ____. To control Fuel Pool water temperature, Augmented Fuel Pool Cooling ____ (2) ____ may be used.

	Temperature Limit	**Augmented Fuel Pool Cooling Mode**
A.	140 degrees F	With Shutdown Cooling
B.	140 degrees F	Without Shutdown Cooling
C.	125 degrees F	With Shutdown Cooling
D.	125 degrees F	Without Shutdown Cooling

Proposed Answer: D

Explanation (Optional):

A. Incorrect: 140 is the temperature limit for the fuel pool cooling demineralizer.

B. Incorrect: 140 is the temperature limit for the fuel pool cooling demineralizer.

C. Incorrect: Augmented Fuel Pool Cooling with Shutdown Cooling Mode 1 can be utilized when the RHR System is performing Shutdown Cooling, the Reactor basin is flooded, and the fuel pool gate is removed.

D. Correct: The design limit is 125 degrees F. With the fuel pool gate installed, Augmented Fuel Pool Without Shutdown Cooling is the only lineup to provide additional cooling.

Technical Reference(s): 2.2.85.2, Augmented Fuel Pool Cooling Mode 2 pages 6 and 7 (Attach if not previously provided)

PNPS 2.2.85, FUEL POOL COOLING AND FILTERING SYSTEM, page 10

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # WTS Bank #2094

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam: 2003

Question Cognitive Level: Memory or Fundamental Knowledge X

Comprehension or Analysis

10 CFR Part 55 Content: 55.41

55.43 7

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	234000 A2.01	
	Importance Rating		3.7

Ability to (a) predict the impacts of the following on the FUEL HANDLING EQUIPMENT; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations: Interlock failure

Proposed Question: SRO Question # 93

Given the following:

- The plant is in Refuel Mode with a core offload in progress
- Two cells were previously emptied and their control rod blades removed and mechanisms withdrawn
- Jumpers have been installed for these control rods to simulate a "Full In" indication in accordance with PNPS 2.2.87.4 JUMPER FOR CONTROL ROD "FULL IN" TO ALLOW MULTIPLE CONTROL ROD REMOVAL DURING AN RFO
- The bridge is currently over the core with the Fuel Grapple in the Normal Up Position.

With these initial conditions the following occurs:

- When the Fuel Grapple is lowered below the Normal Up position, the operator notes that the bridge display does not indicate that a rod block has actuated.
- When questioned, the Control Room reports that the Rod Block Alarm has not annunciated.

(1) What is the significance of these indications

AND

(2) What is the impact on continued refueling activities?

A. (1) The refueling interlocks are inoperable.
(2) Fuel offloading activities must be suspended immediately.

B. (1) The refueling interlocks are inoperable.
(2) Fuel offloading activities can continue for an additional hour while a control rod block is inserted. If the control rod block is not inserted within the hour, fuel offloading activities must be suspended immediately.

C. (1) The refueling interlocks remain operable provided the interlock light illuminates when the Grapple is loaded.

(2) Fuel offloading activities can continue provided that the most recent PNPS 8.10.1, REFUELING PLATFORM INTERLOCKS FUNCTIONAL TEST, is reviewed and the Grapple Loaded Rod Block was demonstrated operable.

D. (1) The refueling interlocks remain operable. The light will not illuminate unless a rod is withdrawn with the bridge over the core and the Grapple is not in the Normal Up Position.
(2) Fuel offloading activities can continue without additional restrictions.

Proposed Answer: A

Explanation (Optional):

A. Correct: Per PNPS 2.2.75, page 65, this is one of the refuel interlocks and is verified via procedure 8.10.1 prior to commencing refuel activities and weekly thereafter. Per TS 3.10.A, fuel offloading must be suspended.

B. Incorrect: Per TS 3.10.A, If one or more required refueling equipment interlocks are inoperable (a) Suspend in-vessel fuel movement with equipment associated with the inoperable interlock(s) immediately.
OR
(b) Insert a control rod withdrawal block AND verify all control rods are fully inserted. Because two rods are withdrawn, option (b) is not a viable option. Additionally there is no allowance for continuing the fuel offload for an hour while the specified action is being performed.

C. Incorrect: This is a required interlock.

D. Incorrect: When the Grapple left the Normal-Up position a rod block should have been generated.

Technical Reference(s): PNPS 2.2.75 FUEL HANDLING AND SERVICING EQUIPMENT, Page 74
Tech Spec Section, 3.10.A (Attach if not previously provided)

PNPS 2.2.75 FUEL HANDLING AND SERVICING EQUIPMENT, Page 65

Proposed References to be provided to applicants during examination: TS 3.10.A

Learning Objective: (As available)

Question Source: Bank #

Modified Bank # (Note changes or attach parent)

New X

Question History:	Last NRC Exam:	
Question Cognitive Level:	Memory or Fundamental Knowledge	X
	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43 7	
Comments:		

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	3	
	Group #		
	K/A #	G 2.1.23	
	Importance Rating		4.4

Ability to perform specific system and integrated plant procedures during all modes of plant operation.

Proposed Question: SRO Question # 94

Given the following conditions:

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

- The boron injection initiation temperature curve has been exceeded
- RPV water level is +5 inches and lowering.
- RPV pressure is being controlled between 900 and 1050 psig with SRVs
- Drywell pressure is 2.9 psig
- Drywell temperature is 145 degrees
- Injection has been terminated and prevented from Condensate and Feedwater, HPCI, RHR and Core Spray.
- MSIVs are closed

Based on the above....

A. IAW EOP-02, RPV Control Failure to Scram, injection from the Condensate and Feedwater systems may be resumed once the APRM downscale lights come in.

B. IAW EOP-03, Primary Containment Control, when Drywell pressure lowers below 2.2 psig reset the scram IAW PNPS 5.3.23, Alternate Rod Insertion.

C. IAW EOP-03, Primary Containment Control, immediately initiate Drywell Spray

D. IAW EOP-02, RPV Control Failure to Scram, injection from the Condensate and Feedwater systems may be resumed once reactor power is reduced to below the boron injection initiation temperature curve.

Proposed Answer: A

Explanation (Optional):

A. Correct – IAW EOP-2, step L-18, injection can be commenced when power is < 3% which correspond to the APRM downscales.

B. Incorrect – resetting the scram is NOT performed IAW EOP-03

C. Incorrect – Conditions have not been met to initiate drywell spray.

D. Incorrect: Level is lowered until the conditions of L-18 are met. Plausible in that it is expected that parameters drop below the curve as power is lowered.

Technical Reference(s): EOP-02, level leg (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: Previous Audit Exam Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G 2.2.19	
	Importance Rating		3.4

Knowledge of maintenance work order requirements.

Proposed Question: SRO Question # 95

Given the following:

- The plant is at rated conditions
- "A" LPCI pump has just tripped during a routine surveillance.
- The cause of the pump trip cannot be immediately determined and corrected.
- All other systems are operable.

Which one of the following is correct regarding the prioritization of the associated work order?

The Shift Manager should characterize the work order as:

A. Priority 1 and direct immediate start of repair efforts in parallel with the initiation and planning of a work order.

B. Priority 1 and direct repair efforts be conducted around the clock following the planning of the work order.

C. Priority 2 and direct repair efforts be conducted around the clock following the planning of the work.

D. Priority 2 and the work week schedule adjusted to accommodate repair. If repairs cannot be completed before exceeding 50% of the allowable LCO time, the priority shall be upgraded to Priority 1.

Proposed Answer: B

Explanation (Optional):

A. Incorrect: This action is ONLY authorized if the maintenance is characterized as Emergency Maintenance. . IAW EN-WM-100, the Shift Manager can then authorize the immediate start of repair efforts, in parallel with initiation and planning of a Priority 1 Work Request/Work Order. The definition of Emergency Maintenance is:

The correction of a condition or deficiency that:

- Constitutes an immediate and direct threat to the health and safety of the public.
- Requires immediate attention to prevent deterioration of plant conditions to a possible unsafe or unstable level, which would then constitute an immediate and direct threat to the health and safety of the public.
- Poses a significant industrial hazard that must be corrected immediately to prevent or mitigate actual serious injury or death.

The pump failure requires entry into an LCO but does not meet the criteria for Emergency Maintenance.

B. Correct: Tech Spec 3.5.A.4 and associated bases requires that LPCI be declared inoperable as all active components are required to operable in order for LPCI to be operable. Per EN-WM-100, Attachment 9.1, a failure or significant degradation with a system that requires entry into a Tech Spec AOT, a Priority 1 work order is required. Per page 7 of EN-WM-100, Priority 1 work orders are to be worked around the clock following the planning of the work order.

C. Incorrect: Per EN-WM-100, Attachment 9.1, a failure or significant degradation with a system that requires entry into a Tech Spec AOT, a Priority 1 work order is required.

D. Incorrect: Per EN-WM-100, Attachment 9.1, a failure or significant degradation with a system that requires entry into a Tech Spec AOT, a Priority 1 work order is required.

Technical Reference(s): EN-WM-100, Work Request (WR) Generation, Attachment 9.1 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank #

Modified Bank #

New X

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge X

	Comprehension or Analysis	
10 CFR Part 55 Content:	55.41	
	55.43	1, 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G 2.3.6	
	Importance Rating		3.8

Ability to approve release permits.

Proposed Question: SRO Question # 96

Following a reactor shutdown the following conditions exist:

- The "A" Miscellaneous Tank is being prepared for overboard discharge
- The Prerelease Permit listed the following Seawater and Salt Service Water Pump Lineup:
    - "A" and "B" Seawater pumps in service
    - "A" and "D" Salt Service Water pumps in service.

While reviewing the Release Permit, the Shift Manager notes that the above lineup has changed to:

- "A" Seawater Pump in service and "B" Seawater Pump secured
- "A" and "E" Salt Service Water pumps in service

Which one of the following is correct regarding the discharge?

The discharge:

A. Is NOT permitted because the "B" Seawater Pump is secured.

B. Is NOT permitted because the Salt Service Water pump configuration has changed.

C. Is permitted because the minimum criterion of at least one Seawater pump in service is satisfied.

D. Is permitted because the minimum criterion of at least one Seawater Pump and two Salt Service Water Pumps is satisfied.

Proposed Answer: A

Explanation (Optional):

A. Correct: The Shift manager is required to verify that the dilution pump combination is the same as on the prerelease report prior to authorizing the discharge. The dilution pump combination is based on the number of SW and SSW pumps running. With the "B" seawater pump secured, the dilution flow used in the prerelease calculations is now less than the existing dilution flow.

B. Incorrect: The release cannot be authorized because the dilution flow assumed has changed in that the "B" Seawater pump is not running. The combination of dilution pumps is based on the number of SSW pumps running and not the specific pumps in service.

C. Incorrect: The release cannot be authorized because the dilution flow assumed has changed in that the "B" Seawater pump is not running.

D. Incorrect: The release cannot be authorized because the dilution flow assumed has changed in that the "B" Seawater pump is not running.

Technical Reference(s): PNPS 7.9.12, LIQUID EFFLUENT RELEASES WITH RETDAS, Attachment 1, sheet 2 of 2. Attachment 2 describes how dilution flow is calculated. (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # TADS 6287

Modified Bank #

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 4

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G 2.4.11	
	Importance Rating	3.5	4.2

Knowledge of abnormal condition procedures.

Proposed Question: SRO Question # 97

With the unit at 55% power, a high Stator Water Cooling temperature condition results in a Turbine Generator run back. The plant is now stable with the following conditions:

- Reactor level and pressure are within their normal ranges
- One Turbine Bypass Valve is OPEN

Which one of the following is required by PNPS 2.4.156, Stator Cooling Water Malfunctions and what is the reason for this action?

A. Lower power using PNPS 2.1.14 Station Power Changes to close the bypass valve.

B. SCRAM the Reactor AND ENTER PNPS 2.1.6, Reactor Scram to close the bypass valve.

C. Raise the Load Limit using PNPS 2.2.99, Main Turbine Generator to restore feedwater temperature to within the limits of 2.4.150, Loss of Feedwater Heating.

D. Raise the Speed Load Changer using 2.1.14 to restore feedwater temperature to within the limits of 2.4.150, Loss of Feedwater Heating.

Proposed Answer: A

Explanation (Optional):

A. Correct – Procedure specifies a power reduction to close the bypass valve

B. Incorrect – Procedure directs that a power reduction be conducted to close the bypass valve

C. Incorrect – Procedure directs that a power reduction be conducted to close the bypass valve. Plausible in that the bypass valve being open is resulting in a partial loss of feedwater heating. If the candidate believes that the load limit is the device that is actuated during a runback, then this action would close the bypass.

D. Incorrect – Procedure specifies a power reduction to close the valve. Raising speed load changer would close the valve but also re-initiate the runback

Technical Reference(s): 2.4.156, Sect. 3.0 & 4.0 (Attach if not previously provided)

Proposed References to be provided to applicants during examination: None

Learning Objective: (As available)

Question Source: Bank # 2009 audit exam number 80

Modified Bank # (Note changes or attach parent)

New

Question History: Last NRC Exam:

Question Cognitive Level: Memory or Fundamental Knowledge

Comprehension or Analysis X

10 CFR Part 55 Content: 55.41

55.43 5

Comments:

Examination Outline Cross-reference:	Level	RO	SRO
	Tier #	2	
	Group #	1	
	K/A #	G 2.3.5	
	Importance Rating		2.9

Ability to use radiation monitoring systems, such as fixed radiation monitors and alarms, portable survey instruments, personnel monitoring equipment, etc.

Proposed Question: SRO Question # 98

The plant was at rated conditions when a major LOCA inside the drywell resulted in fuel damage.

The following indications were noted on the Containment Radiation Monitors:

- At time 00:00
  - Both Torus Monitors, RIT 1001-607A and B, indicate 3 R/hr
  - Both Drywell Monitors, RIT 1001-606A and B indicate 30 R/hr
- At Time 00:15
  - Both Torus Monitors, RIT 1001-607A and B, indicate 7 R/hr
  - Both Drywell Monitors, RIT 1001-606A and B indicate 540 R/hr
- At Time 00:45
  - Both Torus Monitors, RIT 1001-607A and B, indicate 95 R/hr
  - Both Drywell Monitors, RIT 1001-606A and B indicate 3600 R/hr

IAW EP-IP-100.1, EMERGENCY ACTION LEVELS (EALs), a Site Area Emergency EAL was first exceeded at time ____ (1) _____ and a General Emergency EAL was first exceeded at time ____ (2) _____.

	(1) **Site Area Emergency**	(2) **General Emergency**
A.	00:00	00:45
B.	00:00	00:15
C.	00:15	00:45
D.	00:45	00:45

Question Cognitive Level:	Memory or Fundamental Knowledge	
	Comprehension or Analysis	X
10 CFR Part 55 Content:	55.41	
	55.43	5

Comments: