U.S. Nuclear Regulatory Commission

Site-Specific SRO Written Examination

Applicant Information

Name:	
Date: 04/20/2010	Facility/Unit: Brunswick Units 1 & 2
Region: I ☐ II ■ III ☐ IV ☐	Reactor Type: W ☐ CE ☐ BW ☐ GE ■
Start Time:	Finish Time:

Instructions

Use the answer sheets provided to document your answers. Staple this cover sheet on top of the answer sheets. To pass the examination you must achieve a final grade of at least 80.00 percent overall, with 70.00 percent or better on the SRO-only items if given in conjunction with the RO exam; SRO-only exams given alone require a final grade of 80.00 percent to pass. You have 8 hours to complete the combined examination, and 3 hours if you are only taking the SRO portion.

Applicant Certification

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

_____
Applicant's Signature

Results

RO/SRO-Only/Total Examination Values	____ / ____ / ____	Points
Applicant's Scores	____ / ____ / ____	Points
Applicant's Grade	____ / ____ / ____	Percent

Written Exam Reference Index

1. 2OP-32, Condensate and Feedwater System Operating Procedure, Attachment 4, Final Feedwater Temperature Vs Power

2. RCIC System Partial P&ID

3. 0EOP-01-UG, User's Guide, Attachment 5, Heat Capacity Temperature Limit

Final Feedwater Temperature Vs Power

NOTE: The reactor thermal power/temperature data in this table are **NOT** intended for determining reactor power level. The data are based on final feedwater temperature (FFWT) operating data scaled to theoretical plant thermal performance and do **NOT** incorporate degraded performance. Reference Precaution and Limitation 3.10.

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FW Temp
100%	431.4	421.4	321.1
99%	430.5	420.5	320.5
98%	429.6	419.6	319.9
97%	428.7	418.7	319.3
96%	427.8	417.8	318.7
95%	426.9	416.9	318.1
94%	426.0	416.0	317.5
93%	425.1	415.1	316.9
92%	424.1	414.1	316.2
91%	423.2	413.2	315.6
90%	422.2	412.2	315.0
89%	421.2	411.2	314.3
88%	420.3	410.3	313.7
87%	419.3	409.3	313.0
86%	418.3	408.3	312.3
85%	417.3	407.3	311.7
84%	416.2	406.2	311.0
83%	415.2	405.2	310.3
82%	414.2	404.2	309.6
81%	413.1	403.1	308.9
80%	412.0	402.0	308.2
79%	411.0	401.0	307.5
78%	409.9	399.9	306.7
77%	408.8	398.8	306.0
76%	407.7	397.7	305.3
75%	406.5	396.5	304.5
74%	405.4	395.4	303.7
73%	404.2	394.2	303.0
72%	403.0	393.0	302.2
71%	401.9	391.9	301.4
70%	400.7	390.7	300.6
69%	399.4	389.4	299.8
68%	398.2	388.2	298.9
67%	396.9	386.9	298.1

ATTACHMENT 4
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Final Feedwater Temperature Vs Power

RX PWR	Nominal FW Temp	Nominal FW Temp Reduced 10°F	110.3°F Reduced FW Temp
66%	395.7	385.7	297.3
65%	394.4	384.4	296.4
64%	393.1	383.1	295.5
63%	391.7	381.7	294.6
62%	390.4	380.4	293.7
61%	389.0	379.0	292.8
60%	387.6	377.6	291.9
59%	386.2	376.2	290.9
58%	384.7	374.7	290.0
57%	383.3	373.3	289.0
56%	381.8	371.8	288.0
55%	380.3	370.3	287.0
54%	378.7	368.7	286.0
53%	377.2	367.2	284.9
52%	375.6	365.6	283.8
51%	373.9	363.9	282.8
50%	372.3	362.3	281.6
49%	370.6	360.6	280.5
48%	368.8	358.8	279.4
47%	367.1	357.1	278.2
46%	365.3	355.3	277.0
45%	363.4	353.4	275.7
44%	361.5	351.5	274.5
43%	359.6	349.6	273.2
42%	357.6	347.6	271.9
41%	355.6	345.6	270.5
40%	353.5	343.5	269.1
39%	351.4	341.4	267.7
38%	349.2	339.2	266.3
37%	347.0	337.0	264.8
36%	344.6	334.6	263.2
35%	342.3	332.3	261.6
34%	339.8	329.8	260.0
33%	337.3	327.3	258.3
32%	334.7	324.7	256.6
31%	332.0	322.0	254.8
30%	329.2	319.2	252.9
29%	326.3	316.3	251.0
28%	323.3	313.3	249.0
27%	320.2	310.2	246.9
26%	317.0	307.0	244.8
25%	313.6	303.6	242.5
24%	310.1	300.1	240.2
23%	306.5	296.5	237.7

RCIC STEAM POT

Partial P&ID (D-02529 Sheet 1)

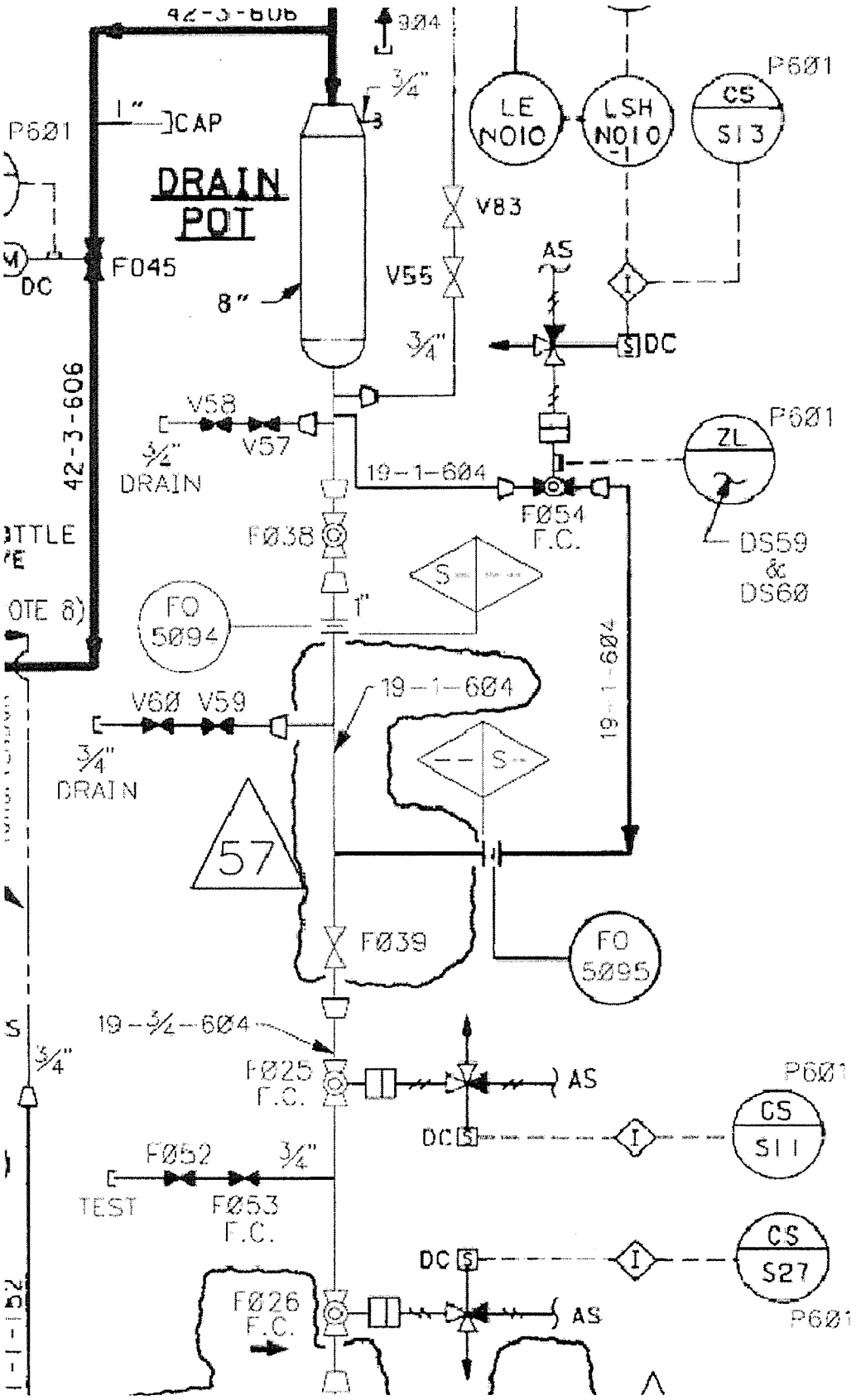
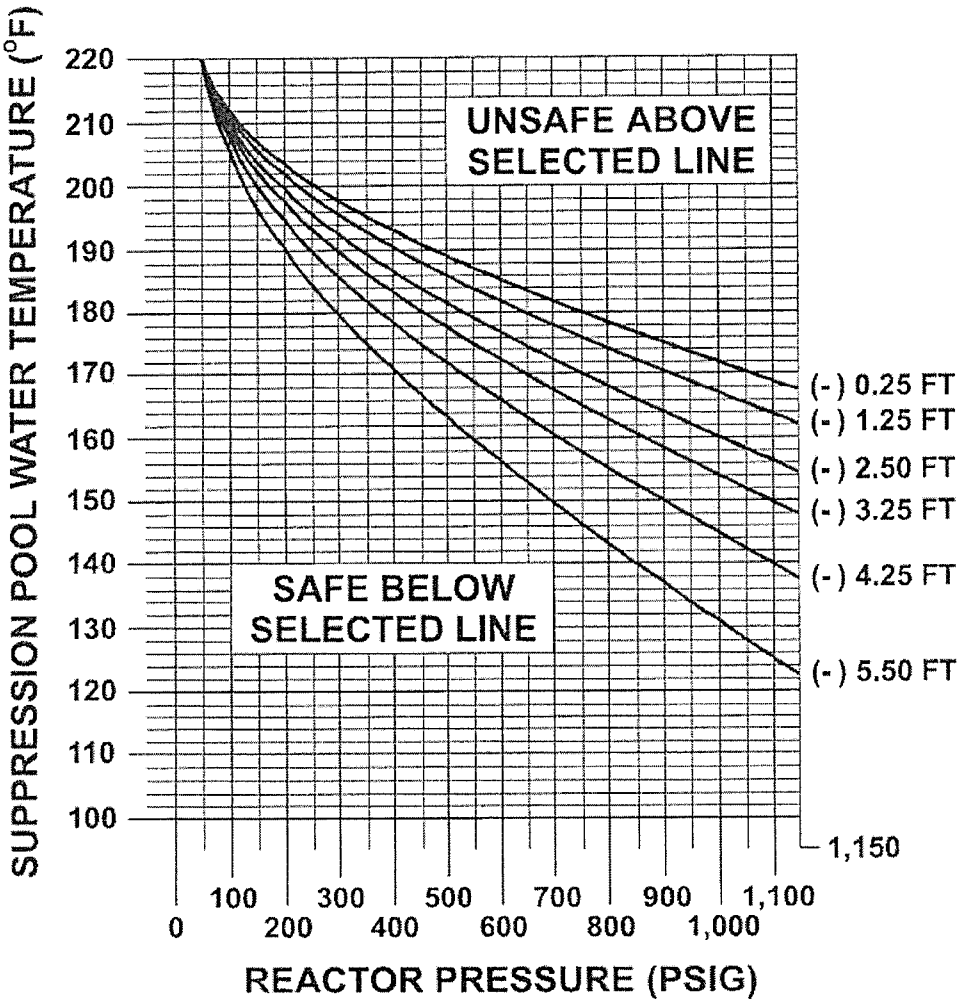


FIGURE 3

Heat Capacity Temperature Limit



SUPPRESSION POOL WATER TEMPERATURE IS DETERMINED BY:

- CAC-TR-4426-1A, POINT WTR AVG **OR**
- CAC-TR-4426-2A, POINT WTR AVG **OR**
- COMPUTER POINT G050 **OR**
- COMPUTER POINT G051 **OR**
- CAC-TY-4426-1 **OR**
- CAC-TY-4426-2

SELECT GRAPH LINE IMMEDIATELY BELOW SUPPRESSION POOL WATER LEVEL AS THE LIMIT.

1. 201001 A1.03 001

Unit Two was operating at rated power when the running CRD Pump tripped, the CRD system is being restarted IAW 2OP-08, Control Rod Drive Hydraulic System Operating Procedure, with the following plant conditions:

2A CRD pump	Running
C12-F002B, Flow Control Valve 2B	Auto
CRD system flow	45 gpm
C12-PCV-F003, Drive Pressure Vlv	Full open

The operator is directed to throttle the C12-PCV-F003 to establish drive water DP between 260 and 275 psid.

Which one of the following choices completes the statement below?

As the operator throttles closed the C12-PCV-F003, the C12-F002B will throttle __(1)__ and drive water DP will __(2)__.

A. (1) open
(2) lower

B. (1) closed
(2) lower

C. (1) open
(2) rise

D. (1) closed
(2) rise

Feedback

K/A: 201001 A1.03
Ability to predict and/or monitor changes in parameters associated with operating the CONTROL ROD DRIVE HYDRAULIC SYSTEM controls including:
CRD system flow
(CFR: 41.5 / 45.5)

RO/SRO Rating:
2.9/2.8

Objective:
CLS-LP-08, Obj 6c
Given plant and CRDHS conditions, predict the values for the following CRDH system parameters:
CRDHS Total System Flow Rate

Reference:
SD-08, CRD Hydraulic System

Cog Level:
Low

Explanation:
With the given conditions (F003 full open) the drive water pressure will be low. The closing of the F003 would reduce the size of the hole in the flowpath thereby raising pressure. With the F002 in auto, it would have to open to maintain the desired flowrate. All of the plausibilities deal with the relationship of the flow control valve to the pressure control valve making any of them possible depending on where the student thinks the valves are.

Distractor Analysis:

Choice A: Plausible because the F002 will open and if the F003 is in a different portion of the flowpath the pressure would drop.

Choice B: Plausible because if the F002 and F003 were in a different alignment in the flowpath this would be possible.

Choice C: Correct Answer, see explanation

Choice D: Plausible because if the F002 was in a different alignment this would be possible, i.e. on the drive water header.

2. 201002 K6.01 001

Which one of the following identifies how the Reactor Manual Control System will be affected by a total loss of the Uninterruptible Power Supply (UPS)?

Control rods ____(1)____.
Control rod position ____(2)____.

A. (1) can be inserted using the Emergency Rod In Notch Override switch
(2) cannot be determined from Full Core Display, 4 Rod group Display, ERFIS, or the process computer

B. (1) cannot be inserted by any method other than scram
(2) cannot be determined from Full Core Display, 4 Rod group Display, ERFIS, or the process computer

C. (1) can be inserted using the Emergency Rod In Notch Override switch
(2) can be determined only from ERFIS or the process computer

D. (1) cannot be inserted by any method other than scram
(2) can be determined only from ERFIS or the process computer

Feedback

K/A: 201002 K6.01
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR MANUAL CONTROL SYSTEM:
Select matrix power
(CFR: 41.7 / 45.7)

RO/SRO Rating:
2.5/2.6

Objective:
CLS-LP-07 Obj 12
List the power supplies to the Reactor Manual Control and Rod Position Indication Systems.

Reference:
SD-52, 120 VAC Electrical System
SD-07, Reactor Manual Control System

Cog Level:
Low

Explanation:
This meets the ka by a loss of UPS which is the power supply to the select matrix and then asking how rods can be moved.
UPS provides power to the select matrix. With no rod being able to be selected, the operator can only insert rods via a scram. UPS also provides power to the Full core and four rod displays, which would be lost. RPIS is also lost. With the rod position indication gone then ERFIS / process computer will display show unknown (??) for each rod.

Distractor Analysis:

Choice A: Plausible because the Emergency In switch bypasses the RMCS logic, but there is no power to select a rod to move. There is no power to the indications (full Core/4 rod) and no RPIS input to the process computer or ERFIS.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the Emergency In switch bypasses the RMCS logic, but there is no power to select a rod to move. ERFIS/process computer has power but the RPIS input is lost.

Choice D: Plausible because ERFIS/process computer has power but the RPIS input is lost.

3. 201006 K5.01 001

Which one of the following defines the purpose of the Rod Worth Minimizer (RWM) IAW Technical Specifications?

A. Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when power is ≥ 19.1%.

B. ✔ Ensures that fuel enthalpy does not exceed 280 cal/gm during a control rod drop accident when reactor power is ≤ 8.75%.

C. Ensures that the MCPR remains ≥1.11, while withdrawing control rods, when power is ≥ 19.1%.

D. Ensures that the MCPR remains ≥1.11, while withdrawing control rods, when reactor power is ≤ 8.75%.

Feedback

K/A: 201006 K5.01
Knowledge of the operational implications of the following concepts as they apply to ROD WORTH MINIMIZER SYSTEM (RWM) (PLANT SPECIFIC) :
Minimize clad damage if a control rod drop accident (CRDA) occurs
P-Spec (Not-BWR6) (CFR: 41.5 / 45.3)

RO/SRO Rating:
3.3/3.7

Objective:
CLS-LP-07.1 Obj. 1
State the purpose of the RWM.

Reference:
Tech Spec 3.3.2.1 Control Rod Block Instrumentation Bases

Cog Level:
Low

Explanation:
OPERABILITY of the RWM, is required in MODES 1 and 2 when THERMAL POWER is ≤ 8.75% RTP. When THERMAL POWER is > 8.75% RTP, there is no possible control rod configuration that results in a control rod worth that could exceed the 280 cal/gm fuel damage limit during a CRDA.
Since the failure consequences for UO2 have shown that sudden fuel pin rupture requires a fuel energy deposition of approximately 425 cal/gm, the fuel damage limit of 280 cal/gm provides a margin of safety from significant core damage which would result in release of radioactivity. Generic evaluations of a design basis CRDA (i.e., a CRDA resulting in a peak fuel energy deposition of 280 cal/gm) have shown that if the peak fuel enthalpy remains below 280 cal/gm

Distractor Analysis:

Choice A: Plausible because the RWM enforces control rod movement from all rods full-in to the Low Power Setpoint (LPSP). (19.1%)

Choice B: Correct Answer, see explanation

Choice C: Plausible because the RWM enforces control rod movement from all rods full-in to the Low Power Setpoint (LPSP). (19.1%) and the RBM ensures MCPR limits.

Choice D: Plausible because the RBM is what ensures the MCPR limits.

4. 203000 K1.02 002

Unit One is operating at rated power with RHR Loop A operating in suppression pool cooling at a flowrate of 11,500 gpm due to entry into PCCP.

A LOCA inside primary containment subsequently occurs. Plant conditions are:

RPV level	Below TAF
RPV pressure	20 psig
Drywell pressure	18 psig

Which one of the following is the expected response of RHR Loop A?

A. Pumping to the reactor only at 11,500 gpm.

B.✓ Pumping to the reactor only at 17,000 gpm.

C. Pumping to the suppression pool only at 11,500 gpm.

D. Pumping to both the reactor and the suppression pool at 17,000 gpm.

Feedback

K/A: 203000 K1.02
Knowledge of the physical connections and/or cause effect relationships between RHR/LPCI: INJECTION MODE (PLANT SPECIFIC) and the following:
Suppression Pool
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating:
3.9/3.9

Objective:
CLS-LP-17 Obj 7
Given plant conditions, determine if the RHR System should automatically initiate in the LPCI mode.

Reference:
sd-17

Cog Level:
High

Explanation: With SPC in operation at 11,500 gpm requires two pump operation. Single pump operation limited to 10,000 gpm. When the Loca Signal occurs the LPCI system will re-aline for injection to the vessel and the SPC flowpath will isolate. Two pump operation will provide 17,000 gpm. The RPV pressure is low enough to provide for the valve manipulations and full flow to the vessel.

Distractor Analysis:

Choice A: Plausible because previous flow was at 11500.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the overrides could have prevented the transfer to the vessel.

Choice D: Plausible because with overrides the SPC valves would not close thus providing two flowpaths.

5. 205000 G2.04.21 002

The following conditions exist on Unit Two following a forced outage shutdown:

Reactor pressure 100 psig
MSIVs closed
RHR Loop B in shutdown cooling with a cooldown in progress
RHR Loop A in standby lineup

Which one of the following describes how the plant will be affected if 2B RPS MG set trips? (assume no operator action)

Reactor pressure will __(1)__ ; a mode change __(2)__ occur.

A. (1) continue to lower
(2) will

B. (1) continue to lower
(2) will not

C. (1) rise
(2) will

D. (1) rise
(2) will not

Feedback

K/A: 205000 G2.04.21
Knowledge of the parameters and logic used to assess the status of safety functions, such as reactivity control, core cooling and heat removal, reactor coolant system integrity, containment conditions, radioactivity release control, etc.
Shutdown Cooling System (RHR Shutdown Cooling Mode)
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating:
4.0/4.6

Objective:
CLS-LP-200-B Obj. 05
Given a set of plant conditions, determine the plant MODE

Reference:
TS

Cog Level:
High

Explanation:
U2 - Loss of RPS MG Set B will cause F008 to close thereby tripping the RHR pump. With a loss of SDC pressure will rise. With the unit at 100 psig the reactor will be in Mode 3 and will stay in mode 3 as the reactor heats up.

Distractor Analysis:

Choice A: Plausible because if there is no effect the pressure will continue to lower and a mode change will occur.

Choice B: Plausible because if there is no effect the pressure will continue to lower and the student could think that a mode change would occur from Mode 4 to 3.

Choice C: Plausible because pressure will rise and the student could think that a mode change would occur from Mode 4 to 3.

Choice D: Correct Answer, see explanation .

6. 206000 K4.02 001

Unit One is at rated power when a steam line break causes the temperatures in the ECCS pipe tunnel to exceed 203°F two minutes ago.

Which one of the following identifies the current status of the Group 4 and Group 5 isolation valves?

A. Group 4 valves closed only.

B. Group 5 valves closed only.

C. Both Group 4 and Group 5 valves closed.

D. Neither Group 4 nor Group 5 valves closed.

Feedback

K/A: 206000 K4.02
Knowledge of HIGH PRESSURE COOLANT INJECTION SYSTEM design feature(s) and/or interlocks which provide for the following:
System isolation: BWR-2,3,4
(CFR: 41.7)

RO/SRO Rating:
3.9/4.0

Objective:
CLS-LP-012-A Obj 6
Given plant conditions, determine if a Group Isolation should occur.

Reference:
SD-12, Primary Containment Isolation System

Cog Level:
Low

Explanation:
RCIC Group 5 has 27 min time delay to allow HPCI isolation to occur and possibly isolate the leak leaving RCIC available.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because it may be thought that the time delay is on the Grp 4 instead of the Grp 5.

Choice C: Plausible if they do not apply any time delays.

Choice D: Plausible if they do apply the time delay to both groups.

7. 209001 K3.01 001

A Dual Unit Loss of Offsite Power occurs with DG1 under clearance and the following electrical plant lineup:

4 kV E-Busses	Energized from their respective available DGs
480 V E-Busses	E5 and E8 only are de-energized

Then a LOCA signal is received on Unit Two.

Which one of the following completes the statement below concerning the ability of Unit Two Core Spray to restore reactor water level?

The Core Spray Pump(s) in ___(1)___ running and injection flow is available through the ___(2)___, Core Spray Inboard Injection Valve.

A. (1) Loop A only is
(2) 2E21-F005A

B. (1) Loop B only is
(2) 2E21-F005B

C✓ (1) both Loops are
(2) 2E21-F005A

D. (1) both Loops are
(2) 2E21-F005B

Feedback

K/A: 209001 K3.01
Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following:
Reactor water level
(CFR: 41.7 / 45.4)

RO/SRO Rating:
3.8/3.9

Objective:
CLS-LP-18 Obj.13b
List the power supplies for each of the following Core Spray System components: MOV's

Reference:
SD-18, Core Spray System

Cog Level:
High

Explanation:
This meets the KA because the power loss is causing a loss of CS and then determining what loops are available to inject (raise reactor water level) based on this loss.
A Core Spray Initiation Signal is present and power is available to both Core Spray pumps, E3 and E4 are energized. With the power loss to E8, MCC 2XD will not have power and the B loop Core Spray valves will be de-energized. So both pumps would be running and injection would only be available from A Loop of Core Spray.

Distractor Analysis:

Choice A: Plausible because loop A pump is running and the injection path is available through A loop. Wrong because the B loop pump is also running.

Choice B: Plausible because the loop pump has power but the injection valve does not. May think that valve power comes from the opposite unit same division similar to the RHR arrangement. The configuration of RHR pumps has power from the opposite unit for the pumps so with a loss of E1 makes a loss of CS pump A plausable.

Choice C: Correct Answer, see explanation

Choice D: Plausible because both pumps do have power but the B injection valve does not. May think that valve power comes from the opposite unit same division like RHR does.

8. 209001 K3.03 001

CS Pump 1A is running for surveillance testing when the CS Pump 1A control power fuses blow inside its 4KV breaker compartment.

A Loss of Off-Site Power then occurs.

Which one of the following choices completes the statement below?

DG1 __(1)__ auto tie onto E1 because the __(2)__.

A. (1) will still
   (2) DG1 output breaker logic will not recognize that the CS Pump 1A breaker is closed

B. (1) will still
   (2) CS Pump 1A breaker will still auto trip open

C. (1) will not
   (2) CS Pump 1A breaker fails to trip open

D. (1) will not
   (2) DG1 output breaker has no control power

Feedback

K/A: 209001 K3.03
Knowledge of the effect that a loss or malfunction of the LOW PRESSURE CORE SPRAY SYSTEM will have on following:
Emergency generators
(CFR: 41.7 / 45.4)

RO/SRO Rating:
2.9/3.0

Objective:
CLS-LP-39 Obj.12
Given plant conditions, determine if permissives are satisfied for the DG output breaker to close.

Reference:
SD-39, Emergency Diesel Generators

Cog Level:
High

Explanation:
This meets the KA because the malfunction (loss of control power to the CS pump) causes the DG not to be able to automatically close onto the E-bus.
CS pump 1A is powered from bus E1 and must load strip prior to EDG #1 O/P breaker closure.

Distractor Analysis:

Choice A: Plausible if examinee believes the blown control power fuse has effected the DG1 output breaker logic

Choice B: Plausible if examinee believes that the blown control power fuse will not prevent the CS Pump 1A breaker from tripping on the LOOP

Choice C: Correct Answer, see explanation

Choice D: Plausible if examinee believes the blown control power fuse has effected the DG1outputt breaker logic

9. 211000 K1.01 001

Which one of the following identifies the relationship between the SLC system and Core Spray Line Break Detection differential pressure instrument?

The ___(1)___ leg of this DP instrument senses ___(2)___ core plate pressure via the SLC/Core Differential Pressure penetration.

A. (1) variable
(2) below

B. (1) variable
(2) above

C. (1) reference
(2) below

D. (1) reference
(2) above

Feedback

K/A: 211000 K1.01
Knowledge of the physical connections and/or cause effect relationships between STANDBY LIQUID CONTROL SYSTEM and the following:
Core spray line break detection: Plant-Specific
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating:
3.0/3.3

Objective:
CLS-LP-18 Obj. 10
Explain the principle of operation of the CS Line Break Detection Instrumentation

Reference:
SD-18, Core Spray System

Cog Level:
Low

Explanation:
This system is comprised of a differential pressure detector which provides Control Room annunciation on detected high DP. The high pressure reference leg of this instrument is exposed to above core plate pressure via the SLC/Core Differential Pressure penetration. The low pressure of this instrument is normally exposed to above core pressure via the Core Spray injection line. This results in the instrument normally measuring core DP (not including core plate DP).

Distractor Analysis:

Choice A: Plausible because the examinee may confuse the reference and variable legs and SLC does discharge below the core plate

Choice B: Plausible because the examinee may confuse the reference and variable legs

Choice C: Plausible because it is the reference leg and SLC does discharge below the core plate.

Choice D: Correct Answer, see explanation

10. 212000 K2.01 001

Which one of the following identifies which RPS MG Set and EPA breakers that trip on a loss of 480 VAC Substation E7?

RPS MG Set __(1)__ EPA breakers __(2)__.

A. (1) A
(2) 1 & 2 only

B. (1) B
(2) 3 & 4 only

C. (1) A
(2) 1 & 2 and alternate source EPA breakers 5 & 6

D. (1) B
(2) 3 & 4 and alternate source EPA breakers 5 & 6

K/A: 212000 K2.01
Knowledge of electrical power supplies to the following:
RPS motor-generator sets
(CFR: 41.7)

RO/SRO Rating:
3.2/3.3

Objective:
CLS-LP-03 Obj 18a
State the power supplies for the following: RPS MG Set A

Reference:
SD-03, Reactor Protection System

Cog Level:
Low

Explanation:
Power for the Motor Generator Sets is tapped off two phases of the normal 480 VAC MC 1CA/1CB (2CA/2CB) power supply for the motor through a stepdown transformer (480V to 120V) from E5/E6 (E7/E8). Selectable reserve power to the Bus is provided from 120 VAC 1E5(2E7) or 1E6(2E8), and is normally selected to Division I. In the event that either RPS M-G Set fails to operate, the alternate power source must be manually selected.
Two EPAs in series are installed downstream of the generator output breaker for each Motor Generator Set and the alternate power supply for the RPS buses. Bus A is protected by EPA-1 and -2; Bus B by EPA-3 and -4. Alternate power is protected by EPA-5 and -6

Distractor Analysis:

Choice A: Plausible because A MG set is lost along with EPA breakers 1 & 2, but these are not the only EPA breakers to trip.

Choice B: Plausible if the examinee picks the wrong power supply and EPA breakers 3 & 4 are powered from RPS MG Set B.

Choice C: Correct Answer, see explanation

Choice D: Plausible if the examinee picks the wrong power supply and EPA breakers 3 & 4 are powered from RPS MG Set B.

11. 214000 A3.04 001

A control rod is notched out from position 12.

The operator observes the 12 indication on the four rod display goes out, comes back on, and then goes out again.

The operator then observes the 13 indication come on and then go out.
No additional rod position is displayed on the four rod display.

Which one of the following identifies the rod position that will be displayed on the RWM? (assume no additional operator action)

A. Position 12 in inverse video.

B. FF with an inferred position of 12.

C. Position 14 in inverse video.

D✓ FF with an inferred position of 14.

Feedback

K/A: 214000 A3.04
Ability to monitor automatic operations of the ROD POSITION INFORMATION SYSTEM including:
RCIS: Plant-Specific
(CFR: 41.7 / 45.7)

RO/SRO Rating:
3.5/3.8

Objective:
CLS-LP-07.1 Obj. 8
Explain how control rod position is inferred and substituted in the RWM

Reference:
SD-07.1, Rod Worth Minimizer System

Cog Level:
Low

Explanation:
For Brunswick the rod control system is RWM which supplies rod blocks and such and indications of the selected rod and position of that rod.
On a rod withdrawal if the even notch position (in this case 14) is failed, as long as RWM detects the previous odd reed switch (13) RWM will provide an inferred position of 14 since RWM also receives data from RMCS that the operator initiated a withdraw motion. If there is an inferred position available, it will not be automatically substituted into RWM. A substitute rod position will be displayed on RWM in inverse video.

Distractor Analysis:

Choice A: Incorrect since 12 will not be displayed but is plausible since position 12 was the last good even position sensed by RWM

Choice B: Incorrect since position 13 was detected but is plausible since no inferred position would be available if 13 failed or if no rod motion command was sensed by RWM and the last sensed reed switch was odd

Choice C: Incorrect since RWM will not automatically substitute inferred position but plausible since the rod is actually at 14, and this would be the display once the operator accepts the inferred position

Choice D: Correct Answer, see explanation

12. 215003 A2.02 001

A plant startup is in progress when the high voltage power supply for IRM G failed low.

All IRMs are on range 1. (Mode switch is in Startup)

Which one of the following choices completes the statements below?

The expected plant response is a ____(1)____ .
The action required IAW the annunciator procedure(s) is to place the ____(2)____.

A. (1) rod block only
(2) joystick on P603 for IRM G to Bypass

B. (1) rod block only
(2) operate switch on the IRM G drawer to STANDBY

C✓ (1) rod block with a half scram
(2) joystick on P603 for IRM G to Bypass

D. (1) rod block with a half scram
(2) operate switch on the IRM G drawer to STANDBY

Feedback

K/A: 215003 A2.02
Ability to (a) predict the impacts of the following on the INTERMEDIATE RANGE MONITOR (IRM) SYSTEM; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
IRM inop condition
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.5/3.7

Objective:
CLS-LP-09.1 Obj. 13c
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: IRM Inop alarm

Reference:
1APP-A-05
SD-9.1, Neutron Monitoring System (SRM/IRM)

Cog Level:
High

Explanation:
A loss of power to the high voltage supply is an Inop trip of the IRM. This will cause a half scram and rod block. In order to clear the cause of this alarm per the annunciator procedure would be to place the IRM in bypass using the joystick on the P603 panel. The question can not state IAW the specific procedure because that would give the answer to the first part of the question.

Distractor Analysis:

Choice A: Plausible because the actions are correct and rod block will occur, but also a half scram will occur.

Choice B: Plausible becasue a rod block will occur and some components placing it in standby will remove the signal from the trip circuit (i.e. standby gas in standby removes the train from the logic).

Choice C: Correct Answer, see explanation

Choice D: Plausible becasue these will occur and some components placing it in standby will remove the signal from the trip circuit (i.e. standby gas in standby removes the train from the logic).

13. 215004 K5.03 001

A plant startup is in progress. A control rod block has occurred. The following nuclear instrument indications are noted:

SRM	Counts	Position
A	$3x10^5$	Full In
B	190	Mid Position
C	$6x10^4$	Full In
D	125	Mid Position

IRM	Counts	Range
A	25/125	3
B	65/125	2
C	35/125	3
D	15/125	3
E	12/125	2
F	55/125	2
G	30/125	3
H	25/125	3

Which one of the following is the minimum required action(s) that will clear the control rod block?

A. Withdrawing SRM A only.

B. Ranging IRM E to range 3.

C. Withdrawing SRM A and C.

D. Ranging IRM B and F to range 3.

Feedback

K/A: 215004 K5.03
Knowledge of the operational implications of the following concepts as they apply to SOURCE RANGE MONITOR (SRM) SYSTEM:
Changing detector position
(CFR: 41.5 / 45.3)

RO/SRO Rating:
2.8/2.8

Objective:
CLS-LP-09.1 Obj. 9a
Describe the insertion/withdrawal of the SRM detectors, including the following:
Reason for maintaining counts between 125 and $2x10^5$.

Reference:
SD-09.1, Neutron Monitoring System (SRM/IRM)

Cog Level:
High

Explanation:
To clear the rod block SRM must be below $2x10^5$ or IRMs must be > range 7. The retract permit is bypassed with IRMs $\geq$ range 3. Withdrawing SRM A will cause the rod block to clear when less than $2x10^5$.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because IRM E is the only Div I IRM below range 3. If all Div I IRMs are above range 3 then the rod block from SRM Retract Permissive in would be bypassed, not the signal from SRM upscale. Also ranging IRM E to range 3 will cause a IRM downscale which is a rod block.

Choice C: Plausible because SRM A does need to be withdrawn and C is above the old setpoint for the upscale alarm. (recent change, old setpoint was $5x10^4$).

Choice D: Plausible because IRM B & F are the only Div II IRMs below range 3 and these do meet the requirements for ranging them to 3. If all Div II IRMs are above range 3 then a rod block from SRM Retract Permissive would be bypassed on Div II, not the signal from SRM upscale.

14. 215005 A2.04 001

Which one of the following identifies the impact a loss of RPS MG Set B will have on the Unit One Power Range Neutron Monitoring system and identifies the action required to energize RPS B from its alternate power supply?

Power will be lost to ____(1)____ 2 and 4.

In order to re-energize RPS B IAW 1OP-03, Reactor Protection System Operating Procedure, the RPS Power Source Select Switch on Panel P610 is required to be placed in the ____(2)____ position.

A. (1) APRMs
   (2) ALT A

B. (1) APRMs
   (2) ALT B

C. (1) Voters
   (2) ALT A

D. (1) Voters
   (2) ALT B

Feedback

K/A: 215005 A2.04
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:
SCRAM trip signals
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.8/3.9

Objective:
CLS-LP-09.6 Obj.12b
Given plant conditions, predict the response of the PRNMS to a malfunction/failure of the following systems/components: 120 VAC Distribution

Reference:
SD-09.6, Power Range Neutron Monitoring System

Cog Level:
High

Explanation:
Each APRM instrument receives power from two power supplies, LVPS 1 and LVPS 4. LVPS 1 is fed from RPS Bus A while LVPS 4 is fed from RPS Bus B. Therefore, a loss of an RPS Bus will not affect operation of the APRM NUMACS. Each of the four VOTERS corresponds to a channel of the A1, A2, B1, and B2 RPS logic. The VOTER outputs to the RPS logic are: A1 (VOTER 1), A2 (VOTER 3), B1 (VOTER 2), and B2 (VOTER 4). Voters 2 and 4 are powered from RPS B. OP-03 contains the steps to re-energize the RPS MG Set in which transferring to alternate power supply can be performed. If this is done then the switch will be placed in Alt B position. Some confusion usually happens as this procedure is performed because the light above the Alt B position is unlit. Students usually think then that Alt B has no power available to energize the RPS Bus and want to take the switch to Alt A which is the energized bus.

Distractor Analysis:

Choice A: Plausible because the APRM lose one power source but have a redundent power supply. The procedure action is plausible because the ALT A is a position switch that is used for transferring the A RPS to alternate. The student may confuse this with transferring to the A RPS power supply because the light will be extinguished above the Alt B position and be on above the Alt A position.

Choice B: Plausible becasue the APRM lose one power source but have a redundent power supply. Alt B is the correct switch position for the transfer switch.

Choice C: The procedure action is plausible because the ALT A is a position switch that is used for transferring the A RPS to alternate. The student may confuse this with transferring to the A RPS power supply because the light will be extinguished above the Alt B position and be on above the Alt A position.

Choice D: Correct Answer, see explanation.

15. 215005 A2.05 001

Unit One is at 94% power when one recirc flow input to APRM 2 fails downscale (zero).

Which one of the following identifies:

(1) the OPRM response to the recirc flow failure and
(2) the required action IAW the annunciator procedures?

A. (1) OPRM 2 only is enabled.
(2) Bypass APRM 2.

B. (1) OPRM 2 only is enabled.
(2) Verify RBM B auto transfers to APRM 3.

C. (1) All OPRMs are enabled.
(2) Bypass APRM 2.

D. (1) All OPRMs are enabled.
(2) Verify RBM B auto transfers to APRM 3.

Feedback

K/A: 215005 A2.05
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:
Loss of recirculation flow signal
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.5/3.6

Objective:
CLS-LP-09.6 Obj 12g
Given plant conditions, predict the response of the PRNMS to a malfunction/failure of the following systems/components: Recirc Flow Module

Reference:
SD-9.6, Power Range Neutron Monitoring System
1APP-A-06

Cog Level:
High

Explanation:
Each Numac processes the signals from one sensor in Loop A and one in Loop B and averages the signals to obtain total recirc flow rate. If one of the two recirc flow signals to an APRM failed to a zero signal with reactor power at 100%, its OPRM becomes enabled because the calculated flow is reduced to one half of its initial value. The other APRM/OPRMs will be unaffected. The RBM has a primary reference from APRM 2 with the primary alternate from APRM 4 and a secondary alternate from APRM 3. Transfer to the alternate alternate requires a critical self test fault on the alternate.
The APP direction is to bypass the effected APRM.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the RBM B will transfer from APRM 2 to its alternate reference which is APRM 4 or its secondary alternate of APRM 3.

Choice C: Plausible because the alarm for OPRM enabled will be in alarm, and the Voters will see the OPRM enabled on all 4 voters for only OPRM 2 though.

Choice D: Plausible because the alarm for OPRM enabled will be in alarm, and the Voters will see the OPRM enabled on all 4 voters for only OPRM 2 though. The RBM B will transfer from APRM 2 to its alternate reference which is APRM 4 or its secondary alternate of APRM 3.

16. 216000 K2.01 001

Which one of the following identifies the power supply to the Unit One RPS A analog trip cabinets?

A. 125 VDC Panel 3A

B. 125 VDC Panel 4A

C. 125 VDC Panel 11A

D. 125 VDC Panel 12A

Feedback

K/A: 216000 K2.01
Knowledge of electrical power supplies to the following:
Analog trip system: Plant-Specific
(CFR: 41.7)

RO/SRO Rating:
2.8/2.8

Objective:
CLS-LP-03 Obj 18h
State the power supplies for the following: Analog Trip System Logic Cabinets

Reference:
SD-03, Reactor Protection System

Cog Level:
Low

Explanation:
There are four cabinets for the RPS, each housing a separate channel (XU-65 through XU-68). Cabinets receive power from DC panels 11A(B) for Unit 1 and DC panels 12A(B) for unit 2. An NLI /Topaz (backup) inverter and a Lambda power supply are located in each cabinet. In order to meet the complete redundancy criteria, the power supplies are designed to be shared in the event of a power supply failure in one cabinet. These four cabinets cause a trip on a loss of power.

Distractor Analysis:

Choice A: Plausible because this is the feed to the Unit 2 Div 1 ESS trip cabinet

Choice B: Plausible because this is the feed to the Unit 1 Div 1 ESS trip cabinet

Choice C: Correct Answer, see explanation

Choice D: Plausible because this is the feed to Unit two analog trip system cabinets.

17. 217000 A1.01 001

Following a loss of feedwater, RCIC automatically initiated and subsequently shut down on high reactor water level.

Current plant status is:
Reactor water level is 170 inches
RCIC flow controller in Auto set at 200 gpm

The RO opens the E51-F045 and then depresses the PF push button on the RCIC flow controller. No other actions are performed.

Which one of the following identifies the indicated flow on the RCIC flow controller that would be observed for these conditions?

A. ✓ 0 gpm

B. 200 gpm

C. 400 gpm

D. 500 gpm

Feedback

K/A: 217000 A1.01
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including:
RCIC flow
(CFR: 41.5 / 45.5)

RO/SRO Rating:
3.7/3.7

Objective:
CLS-LP-016-A Obj.12a
Given plant conditions, with RCIC controlled from the RTGB, determine if the following automatic actions should occur: RCIC System Initiation

Reference:
SD-16, Reactor Core Isolation Cooling System (RCIC)

Cog Level:
High

Explanation:
The RCIC Injection Valve will automatically open upon receiving a low reactor water level signal (LL2) provided that neither the Turbine Trip and Throttle Valve, E51-V8, nor the Turbine Steam Supply Valve, E51-F045, is full closed. Located on the controller face is a PF (programmable function) pushbutton. When depressed an automatic transfer from MANUAL to AUTOMATIC at a predetermined setpoint of 400 GPM will result. NOTE: This button (PF) has no function if the controller is already in AUTOMATIC. Since reactor level is above the LL2 setpoint, and the LL2 initiation relays do not seal in, the injection valve F013 will not auto open. The turbine will develop speed and discharge pressure causing the min flow valve to open, but this flow will not be reflected on the flow indication since the flow instrument is located downstream of the min flow line.
An operator that does not have knowledge of the interlocks associated with the RCIC injection valve, or who has misconceptions about the operation of the RCIC flow controller could easily choose any of the incorrect choices.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this would be correct if the operator also opened F013, or if reactor level dropped below LL2.

Choice C: Plausible because this would be correct if the operator also opened F013, or if reactor level dropped below LL2, and if the operator placed the controller in Manual prior to depressing the PF push button.

Choice D: Plausible because the PF push button would raise RCIC flow to rated (400 gpm) and not maximum per procedure (500 gpm). Achieving 500 gpm would require the flow control setpoint to be manually raised.

18. 217000 A1.04 001

Given the following plant conditions with RCIC in pressure control mode:

RCIC controller output	70%
Bypass to CST Vlv, E51-F022	Throttled
RCIC Flow	300 gpm
RPV pressure	990 psig, slowly rising
RCIC controller	Automatic set @ 300 gpm

Which one of the following identifies two independent actions that will stabilize RPV pressure?

The RO can throttle the E51-F022 in the ____(1)____ direction, or by ____(2)____ the RCIC Flow Controller auto setpoint.

A. (1) open
(2) lowering

B. (1) open
(2) raising

C. (1) closed
(2) lowering

D. (1) closed
(2) raising

Feedback

K/A: 217000 A1.04
Ability to predict and/or monitor changes in parameters associated with operating the REACTOR CORE ISOLATION COOLING SYSTEM (RCIC) controls including:
Reactor pressure
(CFR: 41.5 / 45.5)

RO/SRO Rating:
3.6/3.6

Objective:
CLS-LP-016-A Obj. 17b
Describe how the following evolutions are performed during operation of the RCIC system:
Adjusting RCIC flow in the reactor pressure control mode.

Reference:
RCIC Hard Card

Cog Level:
High

Explanation:
There are two ways to reduce the RPV pressure with the conditions given. One way is to close the 22 valve, thereby decreasing the size of the hole and forcing the turbine to work harder to deliver the same flowrate. The second is to raise the controller setpoint thereby causing the turbine to work harder by forcing more flow through the same size hole..

Distractor Analysis:

Choice A: Plausible becasue these are the opposite of the actual answers and if the operator was trying to raise RPV pressure this would be correct.

Choice B: Plausible because raising is correct and the operator could have a misconception about the 22 valve.

Choice C: Plausible because closing the 22 is correct and the operator could have a misconception about the flow controller.

Choice D: Correct Answer, see explanation.

19. 218000 K3.01 002

Unit One is operating at power with CS pump 1B under clearance.
A small break LOCA occurs simultaneously with a Loss of Off-site Power to both units.

Only DG2 and DG3 start and tie onto their respective E bus.

The following plant conditions exist on Unit One:

AUTO DEPRESS TIMERS INITIATED	In alarm
REACTOR LOW WTR LEVEL INITIATION	In alarm
RPV pressure	600 psig
Drywell pressure	13 psig

Based on the conditions above, which one of the following:

(1) identifies the status of ADS and
(2) how RPV water level restoration is established?

A. (1) ADS will auto initiate.
(2) Level will be recovered with RHR Loop A.

B. (1) ADS will auto initiate.
(2) Level will be recovered with RHR Loop B.

C✓ (1) ADS will not auto initiate.
(2) When ADS is manually initiated level will be recovered with RHR Loop A.

D. (1) ADS will not auto initiate.
(2) When ADS is manually initiated level will be recovered with RHR Loop B.

Feedback

K/A: 218000 K3.01
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 (CFR: 41.7 / 45.4)

RO/SRO Rating:
4.4/4.4

Objective:
CLS-LP-20 Obj. 16b
Given plant conditions, predict how the following will be affected by a loss or malfunction of ADS/SRVs:
b. Reactor water level

Reference:
SD-20, Automatic Depressurization System (ADS)

Cog Level:
High

Explanation:
With the loss of offsite power and 1B CS pump under clearance this would leave only one pump available in each RHR loop. Therefore ADS logic is lost. Level will continue to lower until the ADS valves are manually opened (emergency depressurization) at which time the running low pressure pumps will be able to add water. Injection would be from the A Loop of RHR as the B Loop injection valves do not have power.

Distractor Analysis:

Choice A: Plausible because ADS does have initiation conditions except that the logic will not have the appropriate pumps lined up for injection.

Choice B: Plausible because ADS does have initiation conditions except that the logic will not have the appropriate pumps lined up for injection. B Loop of RHR does not have power to the injection valves

Choice C: Correct Answer, see explanation.

Choice D: Plausible because ADS will not auto initiate but the B Loop of RHR does not have power to the injection valves.

20. 223002 A3.03 001

Reactor Recirculation pumps have tripped due to a low level condition.

G31-F001, RWCU Inboard Isol Vlv, is Closed.
G31-F004, RWCU Outboard Isol Vlv, is Open.

Which one of the following identifies what the Group 3 Isolation Status Box on ERFIS will display in five minutes?

A. ✓ A green GROUP ISOL

B. A red NO GROUP ISOL

C. A yellow GROUP ISOL CMND

D. A green NO GROUP ISOL CMND

Feedback

K/A: 223002 A3.03
Ability to monitor automatic operations of the PRIMARY CONTAINMENT ISOLATION SYSTEM/NUCLEAR STEAM SUPPLY SHUT-OFF including:
SPDS/ERIS/CRIDS/GDS: Plant-Specific
(CFR: 41.7 / 45.7)

RO/SRO Rating:
2.5/2.8

Objective:
CLS-LP-060-A Obj 4d
Describe the methods used to do the following on the ERFIS/SPDS Computer:
Obtain Group Isolation status including valve position

Reference:
SD-60, ERFIS Data Aquisition, Processing and Display

Cog Level:
Low

Explanation:
ERFIS relies on the isolation signal to determine if an isolation is required. Since RWCU did receive a signal, ERFIS will recognize a valid isolation signal with at least one valve closed in the penetration path and remain Green and display GROUP ISOL.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because this is what would be expected with an isolation signal and no valves closed.

Choice C: Plausible because the isolation signal and valve closure time has not expired and can be confused with an incomplete isolation of the penetration flow path (both valves not closed).

Choice D: Plausible because the candidate does not recognize Recirc pump trip is LL2 (same as RWCU) would be indicated if no isolation signal present.

21. 230000 A1.06 001

Given the following small break LOCA conditions on Unit Two:

Drywell pressure	14.8 psig
Suppression chamber pressure	12.5 psig

Which one of the following identifies the response of suppression pool water level after initiating suppression pool sprays?

The CAC-LI-4177, Suppression Pool Level, indication will ___(1)___ slightly due to the ___(2)___ DP between the drywell and suppression pool.

A. (1) lower
(2) higher

B. (1) lower
(2) reduced

C. (1) rise
(2) higher

D. (1) rise
(2) reduced

Feedback

K/A: 230000 A1.06
Ability to predict and/or monitor changes in parameters associated with operating the RHR/LPCI: TORUS/SUPPRESSION POOL SPRAY MODE controls including:
Suppression pool level
(CFR: 41.5 / 45.5)

RO/SRO Rating:
3.3/3.3

Objective:
N/A

Reference:
SD-04, Primary Containment

Cog Level:
High

Explanation:
With the SP at 12.5 psig and then sprays initiated the pressure will lower in the SP and this will cause the higher delta pressure between the DW and SP to force some water down the downcomers to slightly raise the water level in the SP due to the Higher dP. The pumps take a suction from the SP and then spray back to the SP.

Distractor Analysis:

Choice A: Plausible because a higher d/p would be developed from the spray initiation, but level would not lower based on dP.

Choice B: Plausible if the student has backward thinking of what is occurring with d/p. Lower pressure is lowering dP.

Choice C: Correct Answer, see explanation

Choice D: Plausible because a lower d/p would cause level to rise but the d/p will increase when sprays are initiated.

22. 234000 A4.02 002

Given the following conditions on Unit One with the Mode Switch in Refuel:

-Control rod 26-27 is withdrawn to position 48 for blade removal IAW 1OP-08, Control Rod Drive Hydraulic System Operating Procedure.

-RWM is in Bypass

Which one of the following completes the statement below if Rod Select power is turned off?

When Rod Select Power is turned back on __(1)__ can be selected and the Withdraw Permissive white light __(2)__ be lit.

A. (1) any control rod
(2) will

B. (1) any control rod
(2) will not

C. (1) only Control Rod 26-27
(2) will

D. (1) only Control Rod 26-27
(2) will not

Feedback

K/A: 234000 A4.02
Ability to manually operate and/or monitor in the control room:
Control rod drive system
(CFR: 41.7 / 45.5 to 45.8)

RO/SRO Rating:
3.4/3.7

Objective:
CLS-LP-07 Obj. 9e
Describe the conditions which will energize the following indicating lights: Withdraw Permissive

Reference:
SD-07, Reactor Manual Control System

Cog Level:
High

Explanation:
The rod select circuit is designed so that the rod selection is locked in unless power is removed from the select panel. If rod select power is turned off and back on with a rod withdrawn, the interlock will allow selection of other rods, but will generate a rod block if the rod selected is different than the withdrawn rod.

Distractor Analysis:

Choice A: Plausible because any rod may be selected but the withdraw permissive light will not be lit.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because it is a misconception that no other rod may be selected. the withdraw permissive light will not be lit.

Choice D: Plausible because it is a misconception that no other rod may be selected.

23. 239002 A4.04 001

The following annunciators are received while performing 0PT-11.1.2, Automatic Depressurization System and Safety Relief Valve Operability Test:

SPTMS DIV I BULK WTR SETPOINT TS1
SPTMS DIV II BULK WTR SETPOINT TS1

Which one of the following identifies the minimum Suppression Pool temperature required to receive the above annunciators?

A. 95°F.

B. 100°F.

C. 105°F.

D. 110°F.

Feedback

K/A: 239002 A4.04
Ability to manually operate and/or monitor in the control room:
Suppression pool temperature
(CFR: 41.7 / 45.5 to 45.8)

RO/SRO Rating:
4.3/4.3

Objective:
CLS-LP-302M Obj. 1
Given plant conditions, determine if the following AOPs should be entered: AOP-30

Reference:
APP UA-12 5-4(5-5)

Cog Level:
Low

Explanation:
This alarm setpoint is 95°F.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because this is a homogeneous setpoint distractor

Choice C: Plausible because this is the setpoint for *SPTMS DIV I BULK WTR TEMP SETPT TMAX*

Choice D: Plausible because this is the setpoint for Boron Injection Initiation Temperature (BIIT)

24. 259001 A2.06 001

Unit Two is operating at rated power. The DFCS control signal input to 2A RFP has been lost. The RO observes the following:

RFP A CONTROL TROUBLE alarm is received
RFPT A Man/DFCS control switch is in DFCS
DFCS Control light for RFP A on XU-1 is out

Which one of the following describes how RFP 2A will respond, and what operator action is required by *RFP A CONTROL TROUBLE*, to adjust the speed of RFP 2A?

RFP 2A speed will ____(1)____.

The operator can control RFP A speed by ____(2)____.

A. (1) drop to the idle speed setpoint
   (2) operating the RFPT A Man/DFCS Raise/Lower control switch

B.✓ (1) remain at the last known demand
   (2) operating the RFPT A Man/DFCS Raise/Lower control switch

C. (1) drop to the idle speed setpoint
   (2) placing the C32-SIC-R601A, RFPT A Sp Ctl, in Manual and adjusting the output demand

D. (1) remain at the last known demand
   (2) placing the C32-SIC-R601A, RFPT A Sp Ctl, in Manual and adjusting the output demand

Feedback

K/A: 259001 A2.06
Ability to (a) predict the impacts of the following on the REACTOR FEEDWATER SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Loss of A.C. electrical power
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.2/3.2

Objective:
CLS-LP-32.3 Obj. 10j
Given plant conditions and one or more of the following events use plant procedures to determine the actions required to control and/or mitigate the consequences of the event:
Loss of signal from the DFCS

Reference:
UA-13 6-5
SD-32.3, RFP Turbine Speed Control

Cog Level:
High

Explanation:
UPS supplies power to the controls.
From OP-32, Section 5.7.2 (Notes)
IF RFPT B(A) *MAN/DFCS* selector switch is in *DFCS,* AND the DFCS control signal subsequently drops below 2450 rpm, OR increases to greater than 5450 rpm, THEN Woodward 5009 digital controls will automatically assume RFPT speed control and maintain current speed. In this condition, the RFPT will only respond to *LOWER/RAISE* speed control switch commands

From APP UA-13 6-5 (RFP A Control Trouble)
IF RFPT 2A *DFCS CTRL* light on RTGB XU-1 is NOT illuminated, THEN attempt to control RFP turbine speed as necessary using the *LOWER/RAISE* speed control switch

Distractor Analysis:

Choice A: Plausible because the woodward manual control signal automatically tracks the DFCS output signal. An operator without this knowledge could believe the RFP speed would drop to minimum woodward control speed with the DFCS control signal failed

Choice B: Correct Answer, see explanation

Choice C: Plausible because the DFCS control signal has failed. with the DFCS Control light out, the RFP is under manual control of the woodward governor and adjusting the output of the individual RFP Speed Controller will have no effect. An operator without understanding of the hierarchy of the RFP control system could believe this choice is correct.

Choice D: Plausible because the DFCS control signal has failed. with the DFCS Control light out, the RFP is under manual control of the woodward governor and adjusting the output of the individual RFP Speed Controller will have no effect. An operator without understanding of the hierarchy of the RFP control system could believe this choice is correct.

25. 259002 K6.04 001

Unit One is operating at rated power when the Feedwater Flow B indicator has drifted upscale.

Which one of the following identifies the effect this failure has on reactor water level and DFCS control? (Assume no operator actions)

A. RPV water level will initially lower and then return to the current setpoint
DFCS transfers to 1-element control

B. RPV water level will initially rise and then return to the current setpoint
DFCS transfers to 1-element control

C. RPV water level will initially lower and then return to the current setpoint
DFCS remains in 3-element control

D. RPV water level will initially rise and then return to the current setpoint
DFCS remains in 3-element control

Feedback

K/A: 259002 K6.04
Knowledge of the effect that a loss or malfunction of the following will have on the REACTOR WATER LEVEL CONTROL SYSTEM:
Reactor feedwater flow input
(CFR: 41.7 / 45.7)

RO/SRO Rating:
3.1/3.1

Objective:
CLS-LP-32.2 Obj 7b
Given plant conditions, determine the response of the DFCS to the following events:
Loss of any feed flow input

Reference:
APP A-07 4-2, *FW CTL SYS TROUBLE*
SD-32.2, Digital Feedwater Control System

Cog Level:
High

Explanation:
DFCS is going to see the false rise in feed flow and compensate with higher steam flow drawing more inventory from the vessel causing level to initially drop.
The following signals are the permissives to operate in 3 element control:

All steam flows (4) outputs are valid (within 10% of avg)
All feed flows (2) outputs are valid (within 10% of both)
Master control station in Automatic
At least one (1) Feed pump control station is in Automatic
Reactor Power is > 20%
Feed flow and Steam flow matched
with the feed flow failure this will transfer to 1 element control. Level will be maintained based on level only

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the examinee may misinterpret the feed flow indication rise as an actual rise in feedflow which would raise vessel level.

Choice C: Plausible because level will ultimately be maintained at the current setpoint but it will transfer to single element control on greater than 10% difference between feed flow signals.

Choice D: Plausible because the examinee may misinterpret the feed flow indication rise as an actual rise in feedflow which would raise vessel level. Plausible because level will ultimately be maintained at the current setpoint but it will transfer to single element control on greater than 10% difference between feed flow signals.

26. 261000 A3.01 001

Unit Two is operating at rated power with Standby Gas Treatment (SBGT) System controls aligned as follows:

Train A in SYST A PREF
Train B in STBY

Drywell cooling is lost and the reactor scrams on high drywell pressure. Reactor water level drops to 130 inches and is now rising.

Given these plant conditions, which one of the following choices predicts the SBGT system flow indications at the XU-51 panel?

SBGT Train A flow (FI-3150-1) is __(1)__ SCFM.

SBGT Train B flow (FI-3151-1) is __(2)__ SCFM.

A. (1) 0
(2) 0

B. (1) 0
(2) ~3300

C. (1) ~3300
(2) 0

D. (1) ~3300
(2) ~3300

Feedback

K/A: 261000 A3.01
Ability to monitor automatic operations of the STANDBY GAS TREATMENT SYSTEM including:
System flow
(CFR: 41.7 / 45.7)

RO/SRO Rating:
3.2/3.3

Objective:
CLS-LP-10 Obj. 4
Given plant conditions determine if SBGTs should have initiated.

Reference:
SD-10, Standby Gas Treatment System
OP-10, Standby Gas Treatment System Operating Procedure

Cog Level:
High

Explanation:
With the B SBGT train in STBY it will not auto start, this is more like an Off position. There is an auto start signal from high DW pressure and flow should be verified to be greater than 3000 scfm. The control room indicators scale is from 0 to 4500 scfm. Normal system flow is ~3300 scfm.

Distractor Analysis:

Choice A: Plausible because if there was not an initiation signal this would be correct.

Choice B: Plausible because there is an initiation signal but only one train will operate. If the examinee thinks only B only will start then it would indicate 3300 scfm.

Choice C: Correct Answer, see explanation

Choice D: Plausible because there is a initiation signal and the examinee may think that both trains would initiate. The standby position is a common misunderstanding, this is actually an OFF position.

27. 262001 K6.01 001

Which one of the following identifies how the "manually initiated, automatically executed fast bus transfer" capability is affected following a loss of 125V DC Panel 9A?

The fast bus transfer will ____(1)____ if attempted for 4 KV Bus 1B.
The fast bus transfer will ____(2)____ if attempted for 4 KV Bus 1C.

A. (1) occur
(2) occur

B✓ (1) occur
(2) not occur

C. (1) not occur
(2) occur

D. (1) not occur
(2) not occur

Feedback

K/A: 262001 K6.01
Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION:
D.C. power
(CFR: 41.7 / 45.7)

RO/SRO Rating:
3.1/3.4

Objective:
CLS-LP-50.1 Obj 7
Given plant conditions, predict the effect a loss of DC control power will have on the 4160 VAC System.

Reference:
OI-50, 125/250 and 24/48 VDC Electrical Load Lists
SD-50.1, 4160 VAC Electrical System

Cog Level:
Low

Explanation:
BOP Bus 1B has AUTO control power transfer capability where 1C and 1D do not.

Distractor Analysis:

Choice A: Plausible because the auto transfer of control power will occur on 1B, but will not on 1C and D. Recent plant mods have removed some of the auto transfer capabilities on some of the DC control power arrangements (E-busses require a manual transfer of control power).

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the examinee may have the logics reversed.

Choice D: Plausible because the auto bus transfer will not occur on 1C and D, but it will on 1B. Recent plant mods have removed some of the auto transfer capabilities on some of the DC control power arrangements.

28. 262001 K6.02 001

Which one of the following completes the statements below if a Loss of Offsite Power (LOOP) occurs on Unit Two with DG4 under clearance?

RHR Pump 2B ____(1)____ lost its power supply.
The 2-E11-F015B, LPCI Inboard Injection Valve, ____(2)____ lost its power supply.

A. (1) has
(2) has

B.✓ (1) has
(2) has not

C. (1) has not
(2) has

D. (1) has not
(2) has not

Feedback

K/A: 262001 K6.02
Knowledge of the effect that a loss or malfunction of the following will have on the A.C. ELECTRICAL DISTRIBUTION:
Off-site power
(CFR: 41.7 / 45.7)

RO/SRO Rating:
3.6/3.9

Objective:
CLS-LP-39 Obj 9c
Describe the effects on the plant if one or more of the EDGs failed to start during the following conditions:
LOOP

Reference:
SD-17, Residual Heat Removal System

Cog Level:
High

Explanation:
B RHR Pump receives power from E4 which is feed by DG #4. since it has failed then E4 would be de-energized and the B RHR Pump would have no power. The B Loop injection valves are powered from the same division, but opposite units E Bus. That would be E2, which does have power for the injection valves and D RHR pump.

Distractor Analysis:

Choice A: Plausible the B Pump has lost power and one would logically think that the B Loop would be powered by the Div II power source, which is correct except that it is from Unit 1 Div II.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the D RHR has not lost power and one would logically think that the B Loop would be powered by the Div II power source, which is correct except that it is from Unit 1 Div II.

Choice D: Plausible because the D RHR has not lost power and the injection valve has not lost power.

29. 262002 G2.04.11 001

Unit Two is operating at rated power when a complete loss of Uninterruptible Power Supply (UPS) occurs.

Which one of the following completes both statements below?

After the loss of UPS, __(1)__ of the symptoms listed in 0AOP-02.0, Control Rod Malfunction/Misposition, will be present.

IAW 0AOP-12.0, Loss of Uninterruptible Power Supply (UPS) the TCC-V117, Main Turbine Oil Coolers TBCCW Outlet Isolation Valve, is required to be throttled __(2)__.

A. (1) some
   (2) opened

B. (1) some
   (2) closed

C. (1) none
   (2) opened

D. (1) none
   (2) closed

Feedback

K/A: 262002 G2.04.11
Knowledge of abnormal condition procedures.
Uninterruptable Power Supply (A.C./D.C.)
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
4.0/4.2

Objective:
CLS-LP-302G Obj. 4a
Given plant conditions and any of the following AOPs, determine the required supplemental actions:
AOP-12.0, Loss of Uninterruptible Power Supply (UPS)

Reference:
0AOP-12.0, Loss of Uninterruptible Power Supply (UPS)

Cog Level:
Low

Explanation:
AOP-12 directs reference to AOP-02. the loss of four rod and the full core display are symptoms for AOP-02. An action in AOP-12 is IF necessary, THEN THROTTLE CLOSED MAIN TURBINE OIL COOLERS TBCCW OUTLET ISOLATION VALVE, TCC-V117, to maintain normal turbine lube oil temperature

Distractor Analysis:

Choice A: Plausible because some of AOP-02 symptoms are present but the TCC valve is closed not opened.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because it is not readily apparent that AOP-02 would be entered, and the TCC valve is throttled closed not open.

Choice D: Plausible because it is not readily apparent that AOP-02 would be entered.

30. 263000 K2.01 001

Which one of the following identifies the power supply to the Main Turbine Emergency Bearing Oil Pump (EBOP)?

A. 125 VDC Division I

B. 125 VDC Division II

C. 250 VDC Division I

D. 250 VDC Division II

Feedback

K/A: 263000 K2.01
Knowledge of electrical power supplies to the following:
Major D.C. loads
(CFR: 41.7)

RO/SRO Rating:
3.1/3.4

Objective:
CLS-LP-26.1 Obj. 5c
Identify the electrical distribution system which powers the following: Emergency Bearing Oil Pump

Reference:
SD-26.1, Main Turbine Lube Oil and Lube Oil Storage and Conditioning Systems

Cog Level:
Low

Explanation:
The emergency bearing oil pump (EBOP) is provided to supply oil to the bearings of the main turbine when all ac power is lost. The EBOP is driven by a dc motor powered from 250 VDC 2(1)B.

Distractor Analysis:

Choice A: Plausible because 125 VDC Div I is a potential DC source for an emergency pump

Choice B: Plausible because 125 VDC Div II is a potential DC source for an emergency pump

Choice C: Plausible because 250 VDC Div 1 is a potential DC source for an emergency pump

Choice D: Correct Answer, see explanation

31. 264000 A1.01 002

During monthly load testing of DG3, engine outlet lube oil temperature stabilizes at 182° F. A manual adjustment to LO-TCV-2077, Lube Oil Automatic Temperature Control Valve, causes DG3 to trip and lockout on high lube oil temperature.

Prior to the lockout being reset, a line break in the drywell results in the following plant conditions:

RPV water level	+40 inches
RPV pressure	430 psig
Drywell pressure	14.5 psig
Bus E3	Energized from Off-Site power

Which one of the following would be the response of DG3?

DG3 will:

A. ✓ auto start and run with no operator action.

B. auto start only after the lockout relay is manually reset.

C. remain tripped because no auto start signal is present.

D. remain tripped because of the high lube oil temperature trip.

Feedback

K/A: 264000 A1.01
Ability to predict and/or monitor changes in parameters associated with operating the EMERGENCY GENERATORS (DIESEL/JET) controls including:
Lube oil temperature
(CFR: 41.5 / 45.5)

RO/SRO Rating:
3.0/3.0

Objective:
CLS-LP-39 Obj. 18
Describe the effect of an auto start signal on a diesel generator that was tripped and locked out.

Reference:
SD-39, Emergency Diesel Generators
OP-39, Emergency Diesel Generators Operating Procedure

Cog Level:
High

Explanation:
Engine lockout is auto reset by any auto start signal. LOCA start signal present based on hi drywell pressure concurrent with low reactor pressure. The hi lube oil temperature trip is bypassed with any auto start signal present.

Distractor Analysis:

Choice A: Correct answer, see explanation

Choice B: Plausible for the examinee to think the lockout must be reset prior to D/G auto start

Choice C: Plausible because the examinee may not diagnose the presence of a LOCA start signal

Choice D: Plausible for the examinee to think the high lube oil temp trip would prevent an engine start for engine protection

32. 264000 K1.07 001

During accident conditions on Unit Two the following sequence of events occur:

Time (seconds)	Event
0	Drywell pressure rises above the scram setpoint
2	Complete Loss of Off-site Power occurs
8	Reactor pressure is 400 psig
10	DGs energize their respective E Buses
15	Reactor water level drops below LL3

Which one of the following identifies the earliest time that the LPCI pumps will auto start?

A. 10 seconds

B. 15 seconds

C. 18 seconds

D. 20 seconds

Feedback

K/A: 264000 K1.07
Knowledge of the physical connections and/or cause effect relationships between EMERGENCY GENERATORS (DIESEL/JET) and the following:
Emergency core cooling systems
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating:
3.9/4.1

Objective:
CLS-LP-17 Obj 07
Given plant conditions, determine if the RHR System should automatically initiate in the LPCI mode.

Reference:
SD-17, Residual Heat Removal System

Cog Level:
High

Explanation:
The RHR System will automatically start in the LPCI mode of operation in response to either of two initiation signals: reactor vessel low level (LL 3) or drywell high pressure coincident with reactor vessel low pressure.
All RHR Pumps automatically start 10 seconds from receipt of the initiation signal if the Emergency busses are energized (off-site power available). If off-site power is not available, the pumps automatically start 10 seconds from the time the Emergency Diesel Generators re-energize the busses.

Distractor Analysis:

Choice A: Plausible because this is when LPCI power becomes available with a LOCA signal present

Choice B: Plausible because this is when the initiation signal is present from LL3.

Choice C: Plausible because this is applying the 10 second time delay from when the initiation signal is present from hi DW pressure and low reactor pressure and would have started the pumps if electrical power was present.

Choice D: Correct Answer, see explanation.

33. 272000 K3.05 002

Unit One is operating with the AOG system bypassed.
The AOG-HCV-102, AOG Bypass Valve, control switch is in Auto.

At 0800 hours the B SJAE Rad Monitor loses power.
At 0820 hours the A SJAE Rad Monitor reaches it's hi hi radiation setpoint due to a valid condition.

Which one of the following identifies the earliest time that the AOG Bypass Valve will auto close?

A. 0800

B. 0815

C. 0820

D. 0835

Feedback

K/A: 272000 K3.05
Knowledge of the effect that a loss or malfunction of the RADIATION MONITORING System will have on following:
Offgas system
(CFR: 41.5 / 45.3)

RO/SRO Rating:
3.5/3.7

Objective:
CLS-LP-11.0 Obj 5c
Explain the effect that a loss/malfunction of the PRM System will have on the following: AOG System

Reference:
SD-11.0, Process Radiation Monitoring System

Cog Level:
High

Explanation:
AOG-HCV-102, AOG Bypass Valve, will auto close on a valid HI-HI, Lo or Inop trip signal from both detectors after a 15 minute time delay.

Distractor Analysis:

Choice A: Plausible because the examinee may believe that a loss of power to one rad monitor will initiate an isolation signal and not remember the 15 minute time delay.

Choice B: Plausible because the examinee may believe that a loss of power to one rad monitor will initiate an isolation signal and remember the 15 minute time delay.

Choice C: Plausible because the examinee may recognize the valid isolation signal but forget the 15 minute time delay.

Choice D: Correct Answer, see explanation.

34. 288000 K1.01 001

Given the following plant conditions after a Loss of Off-Site Power to Unit One:

DG1	Running at 3575 KW load
DG2	Running at 3680 KW load
RB HVAC	Isolated

The operator is directed to restart Reactor Building HVAC using three (3) supply fans (75 KW each) and three (3) exhaust fans (45 KW each).

Which one of the following identifies the impact on Diesel Generator loading if:

two supply fans and two exhaust fans are sequentially started from MCC 1XG
and
one supply fan and one exhaust fan are sequentially started from MCC 1XH?

A. DG1 only maximum load will be exceeded.

B. DG2 only maximum load will be exceeded.

C. DG1 and DG2 maximum load will be exceeded.

D. DG1 and DG2 will remain within maximum load limits.

Feedback

K/A: 288000 K1.04
Knowledge of the physical connections and/or cause effect relationships between PLANT VENTILATION SYSTEMS and the following:
AC Electrical
(CFR: 41.2 to 41.9 / 45.7 to 45.8)

RO/SRO Rating:
2.6/2.6

Objective:
CLS-LP-39 Obj. 17a
Given plant conditions, OP-39, OP-50.1, AOP-36.2, and/or ASSD procedures, determine the limits for the following DG parameters: Generator kW.

Reference:
AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

Cog Level:
High

Explanation:
Max loading during LOOP is 110% of rated (3500KW x 110% = 3850KW). 2 sets from MCC 1XG (DG1) adds 240 kw for total of 3815 KW. 1 set from MCC 1XH (DG2) adds 120 KW for total of 3800 KW.

Distractor Analysis:

Choice A: Plausible if the examinee adds the wrong values for the fans, supply vs exhaust, then this answer would be correct.

Choice B: Plausible if the examinee thinks that 1XG is from DG2, then this answer would be correct.

Choice C: Plausible if the examinee considers the rated value instead of the emergency value for the answer.

Choice D: Correct Answer, see explanation.

35. 290002 G2.04.46 002

Unit Two is at rated power when the following alarm is received:

RPV FLANGE SEAL LEAK

Which one of the following identifies:

(1) the O-ring that has failed and
(2) the type of instrumentation used to detect the failure?

A. (1) outer
(2) pressure switch

B. (1) outer
(2) temperature switch

C. (1) inner
(2) pressure switch

D. (1) inner
(2) temperature switch

Feedback

K/A: 290002 G2.04.46
Ability to verify that the alarms are consistent with the plant conditions.
Reactor Vessel Internals
(CFR: 41.10 / 43.5 / 45.3 / 45.12)

RO/SRO Rating:
4.2/4.2

Objective:
CLS-LP-01 Obj 3
Describe the plant conditions that could cause RPV Flange Seal Leak (Annunciator A-02, window 5-6) to annunciate.

Reference:
SD-01.2, Reactor Vessel Instrumentation

Cog Level:
High

Explanation:
The head to vessel flange seal is made up by two concentric o-rings that are installed in grooves. A drilled passage connects the annulus between the two o-rings to a pressure switch. If the pressure rises to 600 psig due to the inner o-ring leaking, an annunciator, A-02 5-6 RPV FLANGE SEAL LEAK, will alarm by the pressure switch closing. Failure of both flange seals is detected by the primary containment leak detection system.

Distractor Analysis:

Choice A: Plausible if examinee has a misconception of the O-ring configuration

Choice B: Plausible if examinee has a misconception of the O-ring configuration.
Plausible because some plant leakage detection systems monitor for system process temperature indications rather than pressure.

Choice C: Correct Answer, see explanation

Choice D: Plausible because some plant leakage detection systems monitor for system process temperature indications rather than pressure.

36. 290003 K4.01 001

Which one of the following valid alarm conditions will result in an auto start of the CREV System?

A. *PROCESS RX BLDG VENT RAD HI-HI*

B. *AREA RAD RADWASTE BLDG HIGH*

C. *REACTOR VESS LO LEVEL TRIP*

D✓ *PRI CTMT PRESS HI TRIP*

Feedback

K/A: 290003 K4.01
Knowledge of CONTROL ROOM HVAC design feature(s) and/or interlocks which provide for the following:
System initiations/reconfiguration: Plant-Specific
(CFR: 41.7)

RO/SRO Rating:
3.1/3.2

Objective:
CLS-LP-37 Obj. 4a
Given plant conditions determine if signals exist that would cause the following to automatically start/open:
Emergency Recirculation Fans

Reference:
SD-37, Control Building HVAC System

Cog Level:
Low

Explanation:
An automatic start signal is initiated by any of the following:
a. Any one of three Area Radiation Monitors
(1) Control Room (Channel 1) 1 mr/hr ± .05mr increasing
(2) Control Building Ventilation Intake (Channel 2 or 3) 7 mr/hr ±.05mr increasing

b. LOCA Signal detected by one of the following:
(1) Reactor Water Low Level 2
(2) Drywell Pressure - High

Distractor Analysis:

Choice A: Plausible because the LOCA signal that does start CREV comes from the same device that initiates the Group 6 isolation and the RB HI HI Rad is an input into the Group 6 isolation logic and not into the start logic for CREV.

Choice B: Plausible because radwaste is located below the main control room and control room rad signals do initiate CREV.

Choice C: Plausible since this is a low level scram setpoint and not the LL2 signal that would initiate CREV.

Choice D: Correct Answer, see explanation

37. 295001 K1.03 002

Which one of the following completes the statement below?

A single recirculation pump trip from rated power will cause the value of Critical Power to __(1)__ and Thermal Limits __(2)__ required to be adjusted IAW Technical Specification 3.4.1, Reactor Coolant System (RCS), for continued power operation. (Assume operation greater than 24 hours)

A. (1) rise
   (2) are

B. (1) rise
   (2) are not

C. (1) lower
   (2) are

D. (1) lower
   (2) are not

Feedback

K/A: 295001 K1.03
Knowledge of the operational implications of the following concepts as they apply to PARTIAL OR COMPLETE LOSS OF FORCED CORE FLOW CIRCULATION :
Thermal limits
(CFR: 41.8 to 41.10)

RO/SRO Rating:
3.6/4.1

Objective:
CLS-LP-106-A*13B
Describe how a change in each of the following affects critical power: Mass flow rate

Reference:
OPS-FUN-LP-104-I (Thermal Limits)
Tech Spec 3.4.1, Reactor Coolant System (RCS)

Cog Level:
Low

Explanation:
Critical Power is equal to the bundle power at which OTB occurs. As actual power goes down, void fraction increases. This causes critical power to lower, although not as significantly as actual power. Since actual power drops further than critical power, the Critical Power Ratio gets larger.
CPR = CP / AP

LCO 3.4.1 states:

Two recirculation loops with matched flows shall be in operation,
OR
One recirculation loop may be in operation provided the following limits are applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;
b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;
c. LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," single loop operation limits specified in the COLR; and
d. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Simulated Thermal Power—High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation.

Distractor Analysis:

Choice A: Plausible if examinee has a misconception of Critical Power and the effects of lower core flow on CP. Part (2) is correct.

Choice B: Plausible if examinee has a misconception of Critical Power and the effects of lower core flow on CP. Plausible if the examinee is not aware of the requirements of TS 3.4.1 or is not familiar with the time requirement for the applicable action statement.

Choice C: Correct Answer, see explanation

Choice D: Plausible if the examinee is not aware of the requirements of TS 3.4.1 or is not familiar with the time requirement for the applicable action statement.

38. 295003 A2.05 001

Both Units were operating at rated power when ALL switchyard PCB position indications turn green.

Diesel Generator status:

DG1	Running loaded
DG2	Under clearance
DG3	Running loaded
DG4	Tripped on low lube oil pressure

Which one of the following identifies the AOP(s) that Unit One and Unit Two are required to perform?

Unit One is required to perform __(1)__.
Unit Two is required to perform __(2)__.

A. (1) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
(2) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

B. (1) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses
(2) 0AOP-36.2, Station Blackout

C. (1) 0AOP-36.2, Station Blackout
(2) 0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

D. (1) 0AOP-36.2, Station Blackout
(2) 0AOP-36.2, Station Blackout

Feedback

K/A: 295003 A2.05
Ability to determine and/or interpret the following as they apply to PARTIAL OR COMPLETE LOSS OF A.C. POWER :
Whether a partial or complete loss of A.C. power has occurred
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.9/4.2

Objective:
LOI-CLS-LP-303-A*001
Given plant conditions and control room indications, determine if AOP 36.2, Station Blackout Procedure, should be entered.

Reference:
AOP-36.2, Station Blackout
AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

Cog Level:
High

Explanation:
This meets the KA because the student will have to determine that all green lights is a LOOP on BOTH Units then determine that neither unit is in Station Blackout since each unit has a running and loaded D/G available.

Switchyard PCB green position indication shows all PCB are OPEN, which indicates Loss of ALL offsite power.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because examinee could misdiagnose unit specific D/G availability

Choice C: Plausible because examinee could misdiagnose unit specific D/G availability

Choice D: Plausible because examinee could confuse a "loss of off-site power on both units" as an entry condition for AOP-36.2 on both units

39. 295004 A1.02 001

Following a loss of feedwater on Unit Two, HPCI and RCIC are being used to restore Reactor water level to the normal band.

Then a loss of Battery Bus 2A-1 and 2A-2 occurs.

Which one of the following completes the statement below due to the conditions above?

__(1)__ continues to operate and if RPV water level reaches 211 inches __(2)__ .

A. (1) HPCI
(2) HPCI will still auto trip

B. (1) HPCI
(2) HPCI will not auto trip

C. (1) RCIC
(2) RCIC will still auto shutdown

D. (1) RCIC
(2) RCIC will not auto shutdown

Feedback

K/A: 295004 A1.02
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF D.C. POWER :
Systems necessary to assure safe plant shutdown
(CFR: 41.7 / 45.6)

RO/SRO Rating:
3.8/4.1

Objective:
CLS-LP-19*26B
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event:Loss of DC power

CLS-LP-16*15E
Given plant conditions, predict the RCIC System response to the following conditions:DC power failure

Reference:
AOP-39.0, Loss of C Power
OI-50, 125/250 and 24/48 VDC Electrical Load Lists

Cog Level:
High

Explanation:
Division I DC is required for HPCI start and operation. A loss of Division I DC will make the RCIC inboard isolation logic inoperable which results in failure of the steam supply valve closure on high vessel level. A loss of Div II DC would make RCIC fail and allows for HPCI vessel high water level trip logic to be partially made up (one out of two).

Distractor Analysis:

Choice A: Plausible because HPCI and RCIC are impacted by a loss of either Division of DC. A loss of Div II DC will allow HPCI to continue to operate and will trip on high vessel water level. RCIC flow would be lost on loss of Div II DC.

Choice B: Plausible because HPCI and RCIC are impacted by a loss of either Division of DC. A loss of Div II DC will allow HPCI to continue to operate and will trip on high vessel water level. RCIC flow would be lost on loss of Div II DC.

Choice C: Plausible because HPCI and RCIC are impacted by a loss of either Division of DC. RCIC would continue to inject but would **not** trip on high vessel water level with a loss of Div I DC. HPCI flow would be lost.

Choice D: Correct Answer, see explanation

40. 295005 A1.07 001

Unit One is operating at rated power with DG1 running loaded for a monthly load test. A fault trips the Main Generator Primary Lockout relay.

BOP Bus 1C fails to transfer on the generator lockout due to failure of the SAT supply breaker to close.

Which one of the following identifies the status of the E1 Bus that would be reported to the CRS?

E1 is energized from:

A. DG1 with off-site power available.

B. both DG1 and from off-site power.

C. DG1 with off-site power unavailable.

D. off-site power with DG1 running unloaded.

Feedback

K/A: 295005 A1.07
Ability to operate and/or monitor the following as they apply to MAIN TURBINE GENERATOR TRIP:
A.C. electrical distribution
(CFR: 41.7 / 45.6)

RO/SRO Rating:
3.3/3.3

Objective:
CLS-LP-27*05
State the effect that actuation of a main generator lockout relay will have on the Main Generator and station loads.

CLS-LP-39 /Objectives 3, 7, 12
Given plant conditions, determine if EDGs will automatically start.
Given plant conditions, determine if:
a. EDG output breaker will trip
b. E Bus Master/Slave breakers will trip with the EDG in manual mode

Given plant conditions, determine if permissives are satisfied for the EDG output breaker to close (either automatically or manually).

Reference:
SD-39, Emergency Diesel Generators, Sections 3.2.4, 3.2.6, 3.2.7, and 3.2.10

Cog Level:
High

Explanation:
Based on the conditions the RO will have to determine the status in order to report to the CRS which meets the monitoring AC electrical on a generator trip.
Generator primary lockout is a loss of off-site power signal to DG auto start logic. All four DGs will auto start. The DG1 auto start signal will trip the DG1 output breaker to allow the DG to transfer from the manual to auto mode of operation (governor, voltage regulator, trip circuits). The DG will then tie back onto the bus once the bus stripped interlock and bus undervoltage interlock is satisfied. Otherwise it will continue to run unloaded. Bus 1C fails to transfer from UAT to SAT on the trip. This results in loss of BOP bus 1C but this bus feeds E2, not E1 so Bus E1 will remain powered from off-site power (BOP bus). DG2 will auto start, tie to bus E2. Bus E1 is being powered from off-site power via BOP bus 1D and the DG1 is running unloaded

Distractor Analysis:

Choice A: Plausible because if the peaking relays on the E Bus actuated and tripped the master/slave breakers. Peaking relays could actuate during a fast transfer with a DG in parallel if the turbine had tripped resulting in a backup lockout rather than a primary.

Choice B: Plausible because this would be the most likely configuration if the turbine tripped resulting in a backup generator lockout instead of a primary lockout (would not produce a LOOP signal).

Choice C: Plausible because since this would be the configuration if the BOP bus that failed to transfer to the SAT was bus 1D rather than 1C.

Choice D: Correct Answer, see explanation

41. 295006 K1.02 002

A scram has occurred on Unit Two with the following control rod positions:

Control Rods	Fully inserted with the following exceptions:
	26-31 02
	26-23 02
	22-35 02
	22-27 02
	18-23 02
	18-39 02
	30-23 02
	30-35 02
	34-31 02

Which one of the following choices completes the statement below?

The reactor __(1)__ remain shutdown under all conditions without boron.
The operator must insert the control rods __(2)__ IAW LEP-02.

A. (1) will
(2) by driving rods using RMCS

B. (1) will
(2) by placing the individual scram test switches to the Scram position

C. (1) will not
(2) by driving rods using RMCS

D. (1) will not
(2) by placing the individual scram test switches to the Scram position

Feedback

K/A: 295006 K1.02
Knowledge of the operational implications of the following concepts as they apply to SCRAM :
Shutdown margin
(CFR: 41.8 to 41.10)

RO/SRO Rating:
3.4/3.7

Objective:

Reference:
OI-37.4, Reactor Vessel Control Procedure Basis Document
LEP-02, Alternate Control Rod Insertion

Cog Level:
Low

Explanation:
The given conditions state that 9 control rods are at position 02. All other control rods are fully inserted.
Per OI-37.4, "Analysis has also shown that the reactor will remain shutdown under all conditions if no more than 10 control rods are withdrawn to or beyond position 02."
These conditions are meet making the correct response to part (1) "The reactor WILL remain shutdown under all conditions without boron".

Step 23 of the Reactor Scram Procedure directs "Alternate Control Rod Insertion" per LEP-02.
The steps in LEP-02 will guide you to Section 1.0 which will direct driving control rods using RMCS.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because attempting to insert control rods using the Scram Test Switches is an option available in LEP-02 applicable to other ATWS conditions.

Choice C: Plausible if the examinee is unsure of the number of allowable rods at 02 to satisfy the "shutdown under all conditions without boron" statement.

Choice D: Plausible if the examinee is unsure of the number of allowable rods at 02 to satisfy the "shutdown under all conditions without boron" statement.
Plausible because attempting to insert control rods using the Scram Test Switches is an option available in LEP-02 applicable to other ATWS conditions.

42. 295008 K1.01 001

Which one of the following completes the statement below?

Excessive moisture carryover is caused by __(1)__ reactor water level and will result in __(2)__.

A. (1) high
(2) increased jet pump vibrations

B. (1) high
(2) increased erosion wear of turbine blades

C. (1) low
(2) reduced Reactor Recirc pump NPSH

D. (1) low
(2) increased erosion wear of turbine blades

Feedback

K/A: 295008 K1.01
Knowledge of the operational implications of the following concepts as they apply to HIGH REACTOR WATER LEVEL :
Moisture carryover
(CFR: 41.8 to 41.10)

RO/SRO Rating:
3.0/3.2

Objective:
CLS-LP-01*08
With regard to moisture carryover:
a. Define the term.
b. Describe how it is affected by reactor water level.
c. Describe the adverse affects.

Reference:
SD-01, Nuclear Boiler System, Section 2.1.14.a

Cog Level:
Low

Explanation:
Moisture carryover is defined as that moisture entrained in the steam exiting the Reactor Pressure Vessel. The amount of carryover is affected by the reactor water level. If the water level is too high, the water draining out of the separators tends to back up resulting in increased moisture out the top of the separators. Too much moisture will overload the steam dryers with a resultant decrease in steam quality exiting the reactor vessel. This decreased steam quality will cause increased erosion of turbine blades.

Distractor Analysis:

Choice A: Plausible if examinee confuses the effects of excessive carryover with the effects of moisture carryunder.

Choice B: Correct Answer, see explanation

Choice C: Plausible if examinee confuses carryover with carryunder and if examinee confuses the effects of excessive carryover with the effects of moisture carryunder.

Choice D: Plausible if examinee confuses carryover with carryunder.

43. 295014 A2.03 001

Unit Two is operating at rated power when the following conditions are observed by the RO:

Core Thermal Power initially drops below and then stabilizes slightly above 100%.
Main Generator electrical output (MWe) lowers.

Which one of the following events caused the parameter changes observed above?

A. A single control rod drop.

B. An open Safety Relief Valve.

C. Reactor Recirculation Pump 2A speed step change of 2%.

D. 4A Feedwater Heater Extraction steam isolation.

Feedback

K/A: 295014 A2.03
Ability to determine and/or interpret the following as they apply to INADVERTENT REACTIVITY ADDITION :
Cause of reactivity addition
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
4.0/4.3

Objective:
CLS-LP-302-M*01c
Given plant conditions, determine if the following Abnormal Operating Procedure(s) (AOPs) should be entered: 0AOP 30.0, Safety/Relief Valve Failures

Reference:
SD-20, Automatic Depressurization System (ADS)

Cog Level:
High

Explanation:
An SRV opening initially reduces reactor pressure (increasing Voids) causing reactor power to lower. The EHC system senses lower PAM pressure and reduces TCV position to restore pressure. This causes reduced steam flow to the Main Turbine and lower Generator MW output throughout. Reduced steam flow to MT causes less extraction steam flow to FWHs, causing reduction in final feedwater temperature to the reactor, which combined with pressure restoration, raises reactor power above the initial power level.

Distractor Analysis:

Choice A: Plausible because a control rod drop does provide positive reactivity addition. Generator MWe would also increase.

Choice B: Correct Answer, see explanation

Choice C: Plausible because 2A RR pump speed rising would provide positive reactivity addition. Generator MWe would also increase.

Choice D: Plausible because extraction steam isolation does cause feedwater temperature reduction positive reactivity addition. Generator MWe would also increase. A FWH tube leak would look similar to the SRV opening which is different from the extraction isolation.

44. 295015 K3.01 002

Which one of the following choices completes both statements IAW LEP-02, Alternate Control Rod Insertion?

The RWM is bypassed using a __(1)__.

The reason that the RWM is bypassed is because the __(2)__.

A. (1) keylock switch
   (2) Emergency Rod In Notch Override switch will not work when an Insert Block exists

B. (1) joystick
   (2) Emergency Rod In Notch Override switch will not work when an Insert Block exists

C. (1) keylock switch
   (2) Mode Switch in Shutdown generates a Control Rod Block

D. (1) joystick
   (2) Mode Switch in Shutdown generates a Control Rod Block

Feedback

K/A: 295015 K3.01
Knowledge of the reasons for the following responses as they apply to INCOMPLETE SCRAM :
Bypassing rod insertion blocks
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.4/3.7

Objective:
CLS-LP-007*02d
State the purpose(s) of the following RWM components: Bypass Switch

Reference:
LEP-02, Alternate Control Rod Insertion

Cog Level:
Low

Explanation:
LEP-02 is the procedure the we use to insert rods that have failed to insert on a scram. Direction is given to bypass the RWM which is accomplished by using a keylock switch in the RWM display console.
The RWM is bypassed to override the RWM Enforced Insert Block, allowing rods to be inserted using the Emergency Rod In Notch Override Switch.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because many other components are bypasses using a joystick controller.

Choice C: Plausible if examinee confuses the "Shutdown" withdraw block with an insert block.

Choice D: Plausible because many other components are bypasses using a joystick controller.
Plausible if examinee confuses the "Shutdown" withdraw block with an insert block.

45. 295016G 2.04.01 002

Control Room evacuation has been directed by the Shift Manager due to toxic gas.

Which one of the following identifies the next required immediate operator action that must be performed IAW 0AOP-32.0, Plant Shutdown From Outside the Control Room, (for Unit One) once the main turbine has tripped?

A. Wait until steam flow is less than 3×10^6 lb/hr, then place the reactor mode switch to Shutdown.

B. ✓ Immediately place the reactor mode switch to Shutdown

C. Trip both Reactor Recirculation Pumps

D. Manually Scram the reactor

Feedback

K/A: 295016G 2.04.01
Knowledge of EOP entry conditions and immediate action steps.
Control Room Abandonment
(CFR: 41.10 / 43.5 / 45.13)
There is no EOP for Control Room Abandonment, AOP-32.0 does have entry conditions and immediate actions. K/A applied to AOP.

RO/SRO Rating:
4.6/4.8

Objective:
CLS-LP-302-E*002
List the Immediate Operator Actions required in accordance with 0AOP-32.0, Plant Shutdown from Outside Control Room.

Reference:
AOP-32.0, Plant Shutdown from Outside Control Room

Cog Level:
High

Explanation:

AOP-32 immediate actions are to be performed in a specific order so as to not challenge equipment.
1. **MANUALLY** SCRAM the reactor.
2. **TRIP** the main turbine.
3. **OBSERVE** auxiliary power transferred to the SAT.
4. Unit 1 only: **PLACE** the Reactor Mode Switch to *SHUTDOWN.*
5. Unit 2 only: **WHEN** steam flow is less than 3 x 106 lb/hr, **THEN PLACE** the Reactor Mode Switch to *SHUTDOWN.*
6. **TRIP** both Reactor Recirculation Pumps.
7. **REDUCE** reactor pressure to approximately 700 psig using the bypass valve opening jack.
8. **WHEN** reactor pressure reaches approximately 700 psig, **THEN PLACE** the control switches for the *INBOARD* and *OUTBOARD MSIVS* to *CLOSE.*
9. **PLACE** Mode Selector Switches for Condensate Booster Pumps in *MAN.*
10. **PLACE** Mode Selector Switches for the Condensate Pumps in *MAN.*
11. **GO TO** 1(2)EOP-01-RSP **AND PERFORM CONCURRENTLY** as many of the actions as possible prior to evacuation.

Distractor Analysis:

Choice A: Plausible because this is a required action and the examinee may confuse the required sequence of actions.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this is a required action and the examinee may confuse the required sequence of actions.

Choice D: Plausible because this is a required action and the examinee may confuse the required sequence of actions.

46. 295018 A1.02 001

A manual reactor Scram was performed on Unit Two following a complete loss of RBCCW.

Which one of the following identifies the allowances for operating a CRD Pump under these conditions without cooling water flow IAW 0AOP-16.0, RBCCW System Failure?

The CRD Pump can be operated:

A. for a maximum of 1.5 minutes

B. for a maximum of 10 minutes

C. for a maximum of 20 minutes

D. Indefinitely

Feedback

K/A: 295018 A1.02
Ability to operate and/or monitor the following as they apply to PARTIAL OR COMPLETE LOSS OF COMPONENT COOLING WATER :
System loads
(CFR: 41.7 / 45.6)

RO/SRO Rating:
3.3/3.4

Objective:
CLS-LP-302-H*012a
Given plant conditions and entry into any of the following AOPs, explain the basis for a specific caution, note, or series of procedure steps: 0AOP-16.0, RBCCW System Failure

Reference:
0AOP-16, RBCCW System Failure

Cog Level:
Low

Explanation:
A loss of RBCCW will result in elevated CRD pump component temperatures and could possibly lead to CRD pump failure. Both pumps should be tripped if a total loss of RBCCW occurs, but may be run for up to 20 minutes without RBCCW Cooling if directed by the SRO for rod insertions or RPV level control.

Distractor Analysis:

Choice A: Plausible because Reactor Recirculation pumps must be shutdown within 90 seconds (1.5 minutes) with a loss of seal injection and seal cooling flow.

Choice B: Plausible because Reactor Recirculation pumps are allowed continued operation for a maximum of 10 minutes with no RBCCW cooling flow.

Choice C: Correct Answer, see explanation

Choice D: Plausible because plant OE has shown that a CRD pump can run indefinitely without RBCCW cooling.

47. 295019G 2.02.37 002

Unit Two is operating at rated power. The Non-Interruptible Air (RNA) header pressure has lowered to 65 psig due to a leak.

Which one of the following predicts how the Outboard MSIVs will be affected if the RNA header pressure remains at this value, including the reason?

The Outboard MSIVs will:

A. remain open due to the Pneumatic Nitrogen System supplying the valve actuators in Mode 1.

B. remain open due to the Backup Nitrogen System automatically re-aligning to the valve actuators.

C. eventually drift closed due to the diminishing pneumatic supply to the valve actuators.

D. immediately close due to an automatic isolation signal.

Feedback

K/A: 295019G 2.02.37
Ability to determine operability and/or availability of safety related equipment.
Partial or Complete Loss of Instrument Air
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating:
3.6/4.6

Objective:
CLS-LP-25*08b
Given plant conditions, predict the effect that the following will have on the Main Steam System: Loss of Reactor Non-interruptible Air (RNA)/PNS/Backup Nitrogen.

Reference:
SD-46, Pneumatic Systems, Section 4.2.4

Cog Level:
Low

Explanation:
The pneumatic sources for the outboard MSIVs are Reactor Building Non-Interruptible Air (RNA) System Division I and Division II. The pneumatic sources for the inboard MSIVs are Pneumatic Nitrogen System (PNS) Division I and Division II when operating in mode 1 and (RNA) System Division I and Division II when shutdown. Unlike the SRVs, a loss of PNS does not result in lining up the BU N2 System to the pneumatic operators. An air accumulator located between the MSIV air operator and the check valves provides backup operating air. The capacity of the accumulator is sufficient for the air operator to exercise the valve through one-half of a cycle (open-to-closed or closed-to-open) should the supply air to the operator be interrupted.

Distractor Analysis:

Choice A: Plausible if examinee confuses Inboard pneumatic supply with Outboard pneumatic supply.

Choice B: Plausible because backup nitrogen is capable of supplying the Inboard valves but not the Outboard.

Choice C: Correct Answer, see explanation

Choice D: Plausible there are automatic component isolations associated with the plant Pneumatic System. This is not one.

48. 295020G 2.04.01 001

Unit Two has entered the RSP due to an inadvertent Group 1 isolation.

RCIC is the only injection source available.
SRVs are being used for pressure control.
Reactor water level is 160 inches and lowering.

Which one of the following is the first Suppression Pool Temperature that requires PCCP entry and which additional EOPs (if any) are required to be entered given these conditions?

A. 95°F
None

B. ✓ 95°F
RVCP

C. 105°F
None

D. 105°F
RVCP

Feedback

K/A: 295020G 2.04.01
Knowledge of EOP entry conditions and immediate action steps.
Inadvertent Containment Isolation
(CFR: 41.10 / 43.5 / 45.13)
The EOPs at Brunswick do not have any immediate operator actions, so the question is written for only the EOP entry conditions.

RO/SRO Rating:
4.6/4.8

Objective:
LOI-CLS-LP-300-L Obj. 2
Given plant conditions, determine if the Primary Containment Control Procedure should be entered

Reference:
PCCP and RVCP

Cog Level:
High

Explanation:
EOP-02-PCCP requires entry at 95 degrees SP temperature or 105 degrees if testing is in progress. RSP requires entry into RVCP if level can not be maintained greater than 170 inches. There are no immediate operator actions for EOP entries.

Distractor Analysis:

Choice A: Plausible because SPT is 95°F which is PCCP entry condition. Level is an entry into RVCP.

Choice B: Correct Answer, see explanation

Choice C: Plausible because SPT at 105°F is PCCP entry condition while testing. Though no testing is in progress. Level is an entry into RVCP.

Choice D: Plausible because SPT at 105°F is PCCP entry condition while testing. Though no testing is in progress.

49. 295021 A2.02 002

Following a loss of normal shutdown cooling, alternate shutdown cooling using SRVs is being performed per 0AOP-15.0, Loss of Shutdown Cooling. Plant conditions are:

Reactor water level	400" (N027A/B)
Reactor pressure	175 psig
Suppr chamber press	0.1 psig
Suppr pool temp	80°F
SRVs	1 open
RHR Pumps	2A injecting to the RPV, 2D in suppression pool cooling

Based on these plant conditions which one of the following is an action that is required IAW 0AOP-15.0?

A. Reduce reactor water level.

B. Open an additional SRV.

C. Reduce RHR injection flowrate.

D. Start 2C RHR Pump.

Feedback

K/A: 295021 A2.02
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING :
RHR/shutdown cooling system flow
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.4/3.4

Objective:
CLS-LP-302-L*01a
Given plant conditions, determine if the following Abnormal Operating Procedure(s) (AOPs) should be entered: AOP-15.0, Loss of Shutdown Cooling

Reference:
0AOP-15, Loss of Shutdown Cooling, Page 2 - Symptoms

Cog Level:
High

Explanation:
With the given conditions, reactor pressure is greater than 164 psig than SP pressure. This requires an additional SRV to be opened to make sure that there is sufficient RHR flow through the core to provide cooling.

Distractor Analysis:

Choice A: Plausible because the student may think that lower level would solve the RHR head pressure concerns.

Choice B: Correct Answer, see explanation

Choice C: Plausible because they may think that RHR is pressurizing the reactor.

Choice D: Plausible because they may think that starting a second pump would allow more flow through the core.

50. 295023 K3.03 001

A spent fuel bundle has been dropped on the refuel floor. The following alarm is received:

PROCESS RX BLDG VENT RAD HIGH

Which one of the following predicts the status of Secondary Containment IAW 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity and the reason?

Secondary Containment __(1)__ in order to __(2)__.

A. (1) has auto isolated and SBGT has auto started
   (2) provide an elevated/filtered release point

B. (1) has auto isolated and SBGT has auto started
   (2) prevent an unmonitored release

C. (1) is required to be manually isolated and SBGT manually started (even though no auto actions have occurred)
   (2) provide an elevated/filtered release point

D. (1) is required to be manually isolated and SBGT manually started (even though no auto actions have occurred)
   (2) prevent an unmonitored release

Feedback

K/A: 295023 K3.03
Knowledge of the reasons for the following responses as they apply to REFUELING ACCIDENTS :
Ventilation isolation
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.3/3.6

Objective:
CLS-LP-109-A*01d
Identify the following as related to a Refueling Accident: Plant design features that mitigate the consequences of the accident.

Reference:
0OI-37.9,

Cog Level:
Low

Explanation:
With the high alarm in there is no auto start signal for SBGT or an isolation signal. The AOP states that if new or spent fuel damage is suspected then isolate secondary containment and start SBGT.

The RB HVAC system discharges to the RB Roof. Even though this is greater than 180 feet up this is not considered an elevated release point. The only elevated release point at Brunswick is the Main Stack (310' chimney).

SBGT is the normal mechanism employed under post transient conditions to maintain reactor building pressure negative with respect to the atmosphere since the exhaust from this system is processed and directed to an elevated release point before being discharged to the environment.

This question requires the operator to have knowledge of the reason for ventilation isolation as related to a refueling accident therefore matches the referenced KA statement.
Distractor Analysis:

Choice A: Plausible because the high high is a secondary containment isolation and SBGT start signal not the high alone.

Choice B: Plausible because the high high is a secondary containment isolation and SBGT start signal not the high alone. The RB HVAC is a monitored release path.

Choice C: Correct Answer, see explanation

Choice D: Plausible because it is required to be isolated and SBGT started. The RB HVAC is a monitored release path.

51. 295024G 2.02.37 001

Following a loss of feedwater, HPCI initiated on low reactor water level then tripped on high reactor water level.
Current plant conditions are:

Reactor water level	180 inches, steady
HPCI TURB TRIP	alarm is sealed in
HPCI TURB TRIP SOL ENER	alarm is sealed in
HPCI Initiation Signal/Reset white light	is LIT
HPCI High Water Level Signal Reset white light	is LIT

If drywell pressure rises to 3.0 psig with the above conditions present which one of the following is the minimum required operator action(s) (if any) to allow HPCI injection to the reactor?

A. No operator action is required.

B. Manually open E41-F006, HPCI Injection Vlv.

C. Depress the High Water Level Signal Reset push button.

D. Depress the High Water Level Signal Reset push button <u>first</u> and then manually open E41-F006, HPCI Injection Vlv.

Feedback

K/A: 295024G 2.02.37
Ability to determine operability and/or availability of safety related equipment.
High Drywell Pressure
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating:
3.6/4.6

Objective:
LOI-CLS-LP-19-A, Obj.3m

Cog Level:
High

Explanation:
A Reactor High Water Level trip is initiated and the signal seals-in when a high level is sensed by two instruments. Once the high level condition clears, the trip is reset by a subsequent Reactor Low Level 2 signal or upon depressing the Panel P601 High Level Trip Reset pushbutton. A High Drywell Pressure Initiation signal will not reset the High Water Level trip.

Distractor Analysis:

Choice A: Plausible because the high water level trip does automatically reset on LL2. If high water level is reset with an initiation signal (Hi DW Press), the system automatically aligns for injection without operator actions.

Choice B: Plausible because the high water level trip does automatically reset on LL2. Injection valve requires active initiation condition + Stop Valve (V8) & Steam Supply Valve (F001) not full closed to automatically open. Relay timing has caused HPCI initiation with injection valve not opening (LER 2-90-015).

Choice C: Correct Answer, see explanation

Choice D: Plausible because high water level does not automatically reset due to Hi DW Press. Injection valve requires active initiation condition + Stop Valve (V8) & Steam Supply Valve (F001) not full closed to automatically open. Relay timing has caused HPCI initiation with injection valve not opening (LER 2-90-015)

52. 295025 A2.05 001

Unit One has been operating at rated power for the last 18 months.
A Loss of Off-site Power (LOOP) occurs and cannot be restored for 3 hours.

The RSP directs the following step:

STABILIZE PRESS BELOW 1050 PSIG WITH ONE OR MORE OF THE FOLLOWING SYSTEMS:

* MAIN TURBINE BYPASS VALVES
* MAIN STEAM LINE DRAINS
* RCIC
* HPCI
* SRV - IF A CONTINUOUS PNEUMATIC SUPPLY IS AVAILABLE USE OPENING SEQUENCE

039

Which one of the following choices completes both statements below?

__(1)__ are able to provide sufficient steam flow to stabilize reactor pressure after the LOOP (within the first 10 minutes).

Two hours later __(2)__ can be used to maintain reactor pressure stable.

A. (1) Only SRVs
   (2) MSL Drains

B. (1) Only SRVs
   (2) HPCI

C. (1) HPCI and RCIC (combined)
   (2) MSL Drains

D. (1) HPCI and RCIC (combined)
   (2) HPCI

Feedback

K/A: 295025 A2.05
Ability to determine and/or interpret the following as they apply to HIGH REACTOR PRESSURE:
Decay heat generation
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.4/3.6

Objective:
CLS-LP-19*22c
Given plant conditions, predict how a loss or malfunction of the HPCI System will affect the following:
Ability to remove decay heat

Reference:
0OI-37.3, Reactor Scram Procedure Basis Document

Cog Level: High

Explanation:
The amount of decay heat added depends on the power history of the reactor and the amount of time since the reactor was shut down. The number of fissions that have occurred determines the number of fission fragments in the core. Initial Decay Heat generation is equivalent to approximately 7% (beyond the capacity of HPCI) of the equilibrium power prior to the scram. 1 hour following the scram, Decay Heat generation is equivalent to approximately 1% power (within the capacity of HPCI and maybe RCIC).

Distractor Analysis:

Choice A: Plausible because only SRVs is correct and the use of MSL drains is desired but is dependent upon Off-site power availability (CWIPs needed to allow the main condenser to be available as a heat sink), Group 1 isolation signal remains due to low condenser vacuum with no 1OP-25 or EOP guidance to bypass and reset the isolation signal. Reopening MSIVs would not be procedurally allowed due to Cond/FW and CW systems not having power.

Choice B: Correct Answer, see explanation

Choice C: Plausible because HPCI and RCIC combined capacity is below 7% and the use of MSL drains is desired but is dependent upon Off-site power availability (CWIPs needed to allow the main condenser to be available as a heat sink), Group 1 isolation signal remains due to low condenser vacuum with no 1OP-25 or EOP guidance to bypass and reset the isolation signal. Reopening MSIVs would not be procedurally allowed due to Cond/FW and CW systems not having power.

Choice D: Plausible because HPCI and RCIC combined capacity is below 7% and decay heat generation lowering is correct.

53. 295026 A1.01 001

An ATWS has occurred on Unit One with the following plant conditions:

Reactor power	3%, slowly lowering
RPV water level	+60 inches, steady
RPV pressure	300 psig
Drywell pressure	3.0 psig
Suppression pool temp	108°F

Which one of the following identifies the minimum RHR logic requirement(s), if any, to place Suppression Pool Cooling in service under the current plant conditions?

Suppression Pool Cooling can be placed in service:

A. without the use of the Think Switch or the 2/3 Core Height LPCI Initiation Override keylock switch.

B. only by placing the Think Switch in Manual.

C. only by placing the Think Switch in Manual <u>first</u> and then placing the 2/3 Core Height LPCI Initiation Override keylock switch in Manual Override.

D. only by placing the 2/3 Core Height LPCI Initiation Override keylock switch in Manual Override <u>first</u> and then placing the Think Switch in Manual.

Feedback

K/A: 295026 A1.01
Ability to operate and/or monitor the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:
Suppression pool cooling
(CFR: 41.7 / 45.6)

RO/SRO Rating:
4.1/4.1

Objective:
LOI-CLS-LP-017-A*009
Given an RHR pump or valve, list the interlocks, permissives and/or automatic actions associated with the RHR pump or valve, including setpoints.

Reference:
1OP-17, RHR System Operating Procedure, Page 282, Attachment 8

Cog Level:
High

Explanation:
Reactor pressure is less than 410# with DW pressure greater than 2# which would be a LOCA signal. With a LOCA signal present, RHR can be placed in SPC with the use of the THINK switch.

Distractor Analysis:

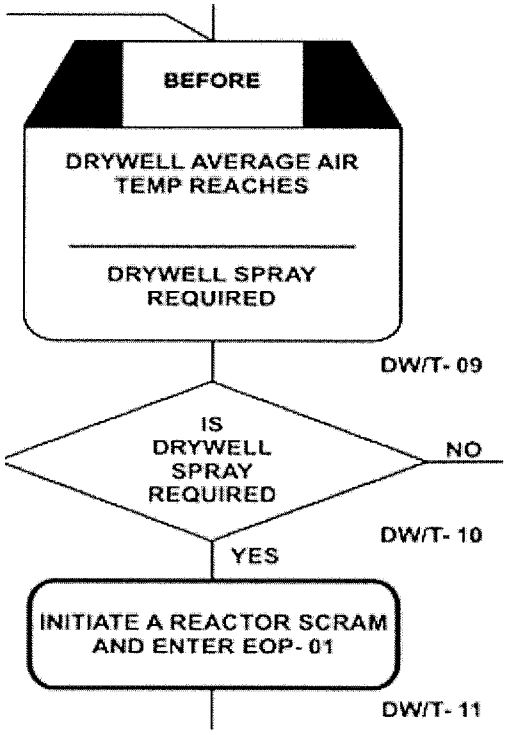
Choice A: Plausible because if the LOCA signal was not present this would be correct.

Choice B: Correct Answer, see explanation

Choice C: Plausible because incorrect recognition of LOCA signal conditions and wrong order of Cooling/Spray logic switch manipulation.

Choice D: Plausible because incorrect recognition of LOCA signal conditions and correct order of Cooling/Spray logic switch manipulation.

54. 295028 K3.05 001



Which one of the following identifies:

(1) the Drywell Average Air Temperature that requires Drywell Sprays in step DW/T-09 and
(2) one of the reasons for initiating a reactor scram at step DW/T-11 IAW 0OI-37.8, Primary Containment Control Procedure Basis Document?

A.✓ (1) 300°F
(2) Operation at power with no recirc pumps operating is not allowed

B. (1) 300°F
(2) when the drywell coolers are secured a high drywell pressure scram signal is imminent

C. (1) 340°F
(2) Operation at power with no recirc pumps operating is not allowed

D. (1) 340°F
(2) when the drywell coolers are secured a high drywell pressure scram signal is imminent

Feedback

K/A: 295028 K3.05
Knowledge of the reasons for the following responses as they apply to HIGH DRYWELL TEMPERATURE :
Reactor SCRAM
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.6/3.7

Objective:
CLS-LP-300-L*005g
Given the Primary Containment Control Procedure, determine the appropriate operator actions if any of the following limits are approached or exceeded: Drywell Design Temperature Limit

Reference:
0OI-37.8, Primary Containment Control Procedure Basis Document, STEPS DW/T-09 through DW/T-17

Cog Level:
Low

Explanation:
A reactor scram is inserted once it has been determined that drywell temperature cannot be maintained below 300°F and DW Spray is required. In order to spray the DW, the Reactor Recirculation Pumps and DW Coolers need to be secured. The reactor is not allowed operation at power without Recirculation Pumps in service. The scram requirement step satisfies shutting down the reactor to support tripping the Recirculation pumps.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because 300°F Drywell temperature is correct. Locking out the DW coolers is an action in the DW spray procedure but is not the reason for scramming the reactor.

Choice C: Plausible because not being able to restore and maintain below 300°F Drywell temperature means exceeding 300°F is allowed without scram but is the step requiring emergency depressurization and tripping the recirc pumps is correct.

Choice D: Plausible because not being able to restore and maintain below 300°F Drywell temperature means exceeding 300°F is allowed without scram but is the step requiring emergency depressurization and Locking out the DW coolers is an action in the DW spray procedure but is not the reason for scramming the reactor.

55. 295030 K2.09 001

The Safety Parameter Display System (SPDS) Plant Status Matrix indicates Suppression Pool level is -31.5 inches.

Which one of the following identifies the color code displayed by SPDS due to Suppression Pool level?

SPDS Suppression Pool level color code is:

A. Green

B. Yellow

C. Red

D. Cyan

Feedback

K/A: 295030 K2.09
Knowledge of the interrelations between LOW SUPPRESSION POOL WATER LEVEL and the following:
SPDS/ERFIS/CRIDS/GDS: Plant-Specific
(CFR: 41.7 / 45.8)

RO/SRO Rating:
2.5/2.8

Objective:
CLS-LP-060*002e
Describe the basic operation of the ERFIS/SPDS Computer: Monitor Display Color Code

Reference:
SD-60 Rev.2, ERFIS DATA ACQUISITION, PROCESSING, AND DISPLAY

Cog Level:
Low

Explanation:
Requires RO to know Tech Spec required Suppression Pool water level of ≥ -31 inches and ≤ -27 inches and that PCCP entry condition is SP level below -31" or above -27" (i.e. -31.2 or -26.8).
Knowing the specific level at which the display turns yellow just informs the operator that it is approaching the High/Low alarm (TS Limits).
SPDS display will be green when SP level is < -27.5" and > -30.5" the indication will turn yellow above -27.5" or below -30.5" until the limit of -27" or -31" is reached at which time the code turns red. The red code alerts the operator of possible PCCP entry condition. AOP-14 must be exited under these conditions.

Distractor Analysis:

Choice A: Plausible because -31.5 is easily confused due to being a negative number which combined with greater than or equal signs make this value within the normal band and therefore Green.

Choice B: Plausible because for the same reason above except approaching alarm limit.

Choice C: Correct Answer, see explanation

Choice D: Plausible because a wrong assumption by the candidate beyond the stem of the question. All of the inputs to SPDS are operable.

56. 295031 K3.03 001

Which one of the following completes the statement below IAW 0EOP-01-UG, User's Guide?

Maintaining reactor water level at the jet pump suction with at least one core spray pump injecting into the reactor vessel at __(1)__ gpm provides assurance that __(2)__ exists.

A. (1) 4700
   (2) adequate core cooling

B. (1) 4700
   (2) the minimum steam cooling water level

C. (1) 5000
   (2) adequate core cooling

D. (1) 5000
   (2) the minimum steam cooling water level

Feedback

K/A: 295031 K3.03
Knowledge of the reasons for the following responses as they apply to REACTOR LOW WATER LEVEL :
Spray cooling
(CFR: 41.5 / 45.6)

RO/SRO Rating:
4.1/4.4

Objective:
CLS-LP-300-B*008
Define all EOP terms per the EOP definitions list in EOP-01-UG.

Reference:
EOP-01-UG, EOP Users Guide

Cog Level:
Low

Explanation:
The reason is adequate core cooling.
Adequate core cooling exists per EOP-UG if RPV level is at the jet pump suction with Core Spray injecting at @ 5000 gpm. Jet pump suction elevation is @ -59", specified in RVCP as -57.5 for instrument readability.

Distractor Analysis:

Choice A: Plausible because 4700 gpm was the old flow requirement prior to EC#63657 and ACC is correct.

Choice B: Plausible because 4700 gpm was the old flow requirement prior to EC#63657 and MSCWL (LL4) is -30 inches (depressurized) and would not be applicable under these accident conditions.

Choice C: Correct Answer, see explanation

Choice D: Plausible because 5000 gpm is correct and MSCWL (LL4) is -30 inches (depressurized) and would not be applicable under these accident conditions.

57. 295032 K2.02 001

During accident conditions on Unit Two SCCP directed restarting Reactor Building HVAC IAW SEP-04, Reactor Building HVAC Restart Procedure.

Shortly following restart of the ventilation system the RO observes the following:

RX BLDG VENT TEMP HIGH	in Alarm
Rx Bldg Vent Exhaust Rad Monitor A indication	2.0 mR/hr
Rx Bldg Vent Exhaust Rad Monitor B indication	3.5 mR/hr

Based on the current conditions which one of the following:
(1) identifies the required action(s) and
(2) the reason for these actions?

A. (1) Continue to operate Reactor Building HVAC
(2) Performance of SEP-04 already defeated all isolation logic.

B. (1) Continue to operate Reactor Building HVAC
(2) To ensure operability of the Rx Bldg Vent Exhaust Rad Monitors.

C. (1) Ensure auto isolation of Reactor Building HVAC
(2) Performance of SEP-04 already defeated all isolation logic.

D. (1) Ensure auto isolation of Reactor Building HVAC
(2) The Rx Bldg Vent Exhaust Rad Monitor readings are no longer reliable.

Feedback

K/A: 295032 K2.02
Knowledge of the interrelations between HIGH SECONDARY CONTAINMENT AREA TEMPERATURE and the following:
Secondary containment ventilation
(CFR: 41.7 / 45.8)

RO/SRO Rating:
3.6/3.7

Objective:
CLS-LP-300-M(K)*011
Given plant conditions involving Reactor Building HVAC system isolation and the Secondary Containment Control Procedure, determine if the Reactor Building HVAC system should be restarted.

Reference:
0EOP-01-SEP-04, Reactor Building HVAC Restart Procedure, Section 2.9

Cog Level:
High

Explanation:
Rx Building Vent Temp Hi alarm indicates temperature in the exhaust duct ≥135°F deg. This exceeds the EQ of the Exh rad monitors. SEP-04 defeats RPV Low level, Hi DW pressure, and Main Stack rad Hi. Rx Bldg Vent Rad Hi-Hi and Vent Temp Hi remain active and should have isolated RBHVAC and started both SBGT trains. SEP-04 also provides verification of these actions should either condition occur.

Distractor Analysis:

Choice A: Plausible because SCCP provided guidance to "restart" RB HVAC which can be interpreted to mean under any conditions. Not ALL isolation logic is bypassed.

Choice B: Plausible because SCCP provided guidance to "restart" RB HVAC which can be interpreted to mean under any conditions. Rad monitor readings not being reliable is correct.

Choice C: Plausible because isolating RB HVAC is correct, but not ALL isolation logic is bypassed.

Choice D: Correct Answer, see explanation

58. 295033 A1.05 002

Following a Reactor Scram on Unit Two due to a loss of Off-site power (LOOP) the following plant conditions exist:

AREA RAD RX BLDG HIGH	In alarm
SOUTH RHR RM FLOOD LEVEL HI	In alarm
SOUTH CS RM FLOOD LEVEL HI	In alarm
Reactor Building 20' Rad Level	Approaching Max Norm Operating Rad
Reactor Building 20' Temperature	Approaching Max Norm Operating Temp

Based on the conditions above which one of the following identifies:
(1) the source of the leak and
(2) the operator action required IAW SCCP?

A. (1) SDV
(2) Open seven ADS valves.

B. (1) RBCCW
(2) Open seven ADS valves.

C. (1) SDV
(2) Cooldown within Technical Specification limits

D. (1) RBCCW
(2) Cooldown within Technical Specification limits

Feedback

K/A: 295033 A1.05
Ability to operate and/or monitor the following as they apply to HIGH SECONDARY CONTAINMENT AREA RADIATION LEVELS :
Affected systems so as to isolate damaged portions
(CFR: 41.7 / 45.6)

RO/SRO Rating: 3.9/4.0

Objective: CLS-LP-300-M*08a
8. Given plant conditions and the Secondary Containment Control Procedure, determine if any of the following are required:
   a. Manual reactor scram
   b. Consider Anticipation of Emergency Depressurization
   c. Emergency Depressurization

Reference:
RSP, SCCP, 0OI-37.9

Cog Level: High

Explanation:This meets the KA due to having to reset RPS to isolate the affected system (SDV leaking) thereby closing the scram valves which are the source of the leak causing the high rad levels in the reactor building.

Reactor Scram due to LOOP providing indications of SDV rupture.
The Maximum Normal Operating Values are the highest radiation levels expected to occur during normal plant operating conditions with all directly associated support and control systems functioning properly. Separate radiation levels are provided for each Secondary Containment area.
Flood level Hi is MNOWL entry condition to SCCP. LOOP automatically provides Groups 1,2,3,6,8, & 10 isolations. No RWCU (Grp 3) isolation failure provided in stem. Based upon flood level status, along with rising 20' temperature and radiation leads to SDV rupture.
2 areas above MNOWL with primary system discharge requires Reactor Scram, cooldown <100 deg/hr, and consideration for anticipation of ED. No areas have reached Max Safe Operating Values, Emergency Depressurization is not required.
RPS can be reset and SCCP directs isolating the primary system discharge, main condenser not available due to LOOP

Distractor Analysis:
Choice A: Plausible because the SDV is correct and if two areas above MSOWL with a primary system leak requires ED.

Choice B: Plausible because RBCCW leak could provide rising room levels, but area Rad and Temperature would not rise and if two areas above MSOWL with a primary system leak requires ED..

Choice C: Correct Answer

Choice D: Plausible because RBCCW leak could provide rising room levels, but area Rad and Temperature would not rise and cooldown w/i TS limits is correct.

SRO Only Basis: N/A

59. 295035 A1.01 002

The unit is operating at rated power when the following alarms are received:

RX BLDG DIFF PRESS HIGH/LOW

Which one of the following completes the statement below?

The RB Supply and Exhaust Fans __(1)__ receive a trip signal and the SBGT System __(2)__ receive a start signal.

A. (1) will
   (2) will

B. (1) will
   (2) will not

C. (1) will not
   (2) will

D. (1) will not
   (2) will not

Feedback

K/A: 295035 A1.01
Ability to operate and/or monitor the following as they apply to SECONDARY CONTAINMENT HIGH DIFFERENTIAL PRESSURE:
Secondary containment ventilation system
(CFR: 41.7 / 45.6)

RO/SRO Rating:
3.6/3.6

Objective:
CLS-LP-037.1*06a
List the signals that will cause the following to automatically stop: Reactor Building supply fans

Reference:
APP UA-12 3-3 and SD-37.1

Cog Level:
Low

Explanation:
Fans trip on excessive building differential pressure (+4 inches or -4 inches) but SBGT does not start (APP UA-12 3-3)

Distractor Analysis:

Choice A: Plausible because RBHVAC will trip is correct and SBGT does not auto start.

Choice B: Correct Answer, see explanation

Choice C: Plausible because this alarm is easily confused with RB Static Press Diff - Low which would not trip RB HVAC and SBGT trains auto starting to restore RB DP could be considered common sense but wrong.

Choice D: Plausible because this alarm is easily confused with RB Static Press Diff - Low which would not trip RB HVAC and SBGT trains auto starting to restore RB DP could be considered common sense but wrong.

60. 295037 K1.03 002

Which one of the following identifies IAW LPC:

(1) the highest SLC tank level that will ensure the reactor is maintained shutdown during hot standby conditions and

(2) the minimum pounds of Borax that will ensure the reactor will remain shut down irrespective of control rod position or reactor temperature?

A. (1) 0%
(2) 2461

B. (1) 32%
(2) 2461

C. (1) 0%
(2) 6080

D. (1) 32%
(2) 6080

Feedback

K/A: 295037 K1.03
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)
(CFR: 41.8 to 41.10)

RO/SRO Rating:
4.2/4.4

Objective:
CLS-LP-005*03
List the positive reactivity effects that must be overcome by SLC injection.

Reference:
0OI-37.5

Cog Level:
Low

Explanation:
The Hot Shutdown Boron Weight (HSBW) is not a quantity which can be measured by the operator. An SLC tank level below 32% or approximately 2461 pounds of borax injected are equivalent to the HSBW. The Hot Shutdown Boron Weight is defined to be the least weight of soluble boron which, if injected into the reactor vessel and uniformly mixed, will maintain the reactor shut down under hot standby conditions. These values are determined in calculation 0EOP-WS-16. It has been decided to use 32% to represent the Hot Shutdown Boron Weight in the procedure. This value can be read by the operator on the indication in the Control Room. The borax concentration for Hot Shutdown Boron Weight is calculated as approximately 2461 pounds.
The Cold Shutdown Boron Weight (CSBW) is not a quantity which can be measured by the operator. An SLC tank level of 0% or 6080 pounds of borax injected are equivalent to the CSBW. The Cold Shutdown Boron Weight is defined to be the least weight of soluble boron which, if injected into the reactor vessel and uniformly mixed, will maintain the reactor shut down under all conditions. This weight is utilized to assure the reactor will remain shut down irrespective of control rod position or reactor temperature.

Distractor Analysis:

Choice A: Plausible because 0% is CSBW and 2461 is HSBW borax concentration

Choice B: Plausible because 32% is correct HSBW and 2461 is HSBW borax concentration

Choice C: Plausible because 0% is CSBW and 6080 is correct for CSBW.

Choice D: Correct Answer, see explanation

61. 295038 K2.10 002

Unit Two startup is in progress with both Mechanical Vacuum Pumps (MVPs) in service establishing main condenser vacuum when the following alarm is received:

MAIN STEAM LINE RAD HI-HI/INOP

The operator determines that MSL A Rad Monitor, D12-RM-K603A, is exceeding the Hi-Hi setpoint and is the only cause for the alarm.

Which one of the following identifies the response of the MVPs?

A. MVP A will trip only.

B. Both MVPs will remain running.

C. Both MVPs will trip and OG-V7, Hogging Valve, will close.

D. Both MVPs will trip and OG-V7, Hogging Valve, will remain open.

Feedback

K/A: 295038 K2.10
Knowledge of the interrelations between HIGH OFF SITE RELEASE RATE and the following:
Condenser air removal.
(CFR: 41.7 / 45.8)

RO/SRO Rating:
3.2/3.4

Objective:
CLS-LP-30*11b
Given the necessary plant conditions, describe the effect that a malfunction or loss of the Condenser Air Removal/Augmented Off-Gas System would have on the following: Radioactive Release Rates

Reference:
SD-30, Condenser Air Removal and Off-gas Recombiner System

Cog Level:
High

Explanation:
Hi-Hi trip on the MSL Rad channels will cause both Mechanical vacuum pumps to trip and OG-V7 valve to close. The logic trips the MVPs when a hi-hi condition is present in each division. The MVPs discharge via the 1.8 minute holdup line to the main stack. MSL rad high conditions directly impact off-site release rates with the MVPs in service due to no discharge path processing. MVPs are only allowed to be operated below 5% reactor power due no hydrogen explosion hazards present.

Distractor Analysis:

Choice A: Plausible due to MSL A Rad Hi-Hi causing MVP A trip (trip in one division trips a MVP).

Choice B: Correct answer, see explanation

Choice C: Plausible due to a single MSL Rad Hi-Hi in one division causing a complete isolation and would be correct if a second MSL Rad HI-Hi were received.

Choice D: Plausible due to a single MSL Rad Hi-Hi in one division causing a complete isolation and would be correct if a second MSL Rad HI-Hi were received and not recognizing trip impact on OG-V7.

62. 300000 K5.13 001

Unit Two is operating at rated power when plugging of C12-D006A, Supply Air Filter, causes the *SCRAM VALVE PIL AIR HDR PRESS HI/LO* alarm to be received.

Which one of the following identifies the impact of lowering scram air header pressure?

The CRD Scram Outlet Valves will fail __(1)__ on a low scram air header pressure and the DW Lower Vent Dampers will reposition to the __(2)__ position.

A. (1) open
   (2) MAX

B. (1) open
   (2) MIN

C. (1) closed
   (2) MAX

D. (1) closed
   (2) MIN

Feedback

K/A: 300000 K5.13
Knowledge of the operational implications of the following concepts as they apply to the INSTRUMENT AIR SYSTEM:
Filters
(CFR: 41.5 / 45.3)

RO/SRO Rating:
2.9/2.9

Objective:
CLS-LP-08 Obj. 7d
State the normal and fail position for the following components: CRD Scram Outlet Valves

Reference:
APP-A-07
SD-08, CRD Hydraulic System

Cog Level:
Low

Explanation:
Loss of the air supply will result in the in-service flow control valve closing. With no drive water pressure RMCS will not be able to move rods but they could still be scrammed. If air pressure would continue to lower below 40 psig the scram inlets and outlet valves would fail open on the loss of air. The lower DW dampers go to the MAX position on a scram as sensed by pressure switches in the scram air header.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible since it does fail open and it may seem correct that more cooling would be needed in the upper part of the DW since hot air rises.

Choice C: Plausible because some valves do fail closed (CRD flow control valve) and the dampers do go to the MAX position

Choice D: Plausible because some valves do fail closed (CRD flow control valve) and it may seem correct that more cooling would be needed in the upper part of the DW since hot air rises.

63. 400000 K4.01 001

Which one of the following choices completes the statements below?

The highest CSW system pressure that will auto start the standby CSW pump is __(1)__ psig.

If pressure remains below this setpoint for at least __(2)__ seconds the SW-V3(V4), SW to TBCCW Hxs Otbd(Inbd) Isol, will reposition to their throttled positions.

A. (1) 65
(2) 30

B. (1) 65
(2) 70

C. (1) 40
(2) 30

D. (1) 40
(2) 70

Feedback

K/A: 400000 K4.01
Knowledge of CCWS design feature(s) and or interlocks which provide for the following:
Automatic start of standby pump
(CFR: 41.7)

RO/SRO Rating:
3.4/3.9

Objective: CLS-LP-43 Obj 6d
Given plant conditions, predict whether any of the following pumps should start: Conventional Service Water Pumps

Reference:
SD-43, Service Water System
AOP-19, Conventional Service Water System Failure

Cog Level:
Low

Explanation:
The CSW pumps will auto start at 40 psig, the RCC pumps start at 65 psig.
The SW-V3/4 throttle to a mid position if the low pressure exists for 70 seconds.
The DG cooling valves swap to the opposite unit after low pressure for 30 seconds.

Distractor Analysis:

Choice A: Plausible because the RCC pumps auto start at 65 psig and the DG cooling valves swap to the opposite unit after low pressure for 30 seconds.

Choice B: Plausible because the RCC pumps auto start at 65 psig.

Choice C: Plausible because the DG cooling valves swap to the opposite unit after low pressure for 30 seconds.

Choice D: Correct Answer, see explanation.

64. 600000 K2.01 001

Which one of the following completes both statements?

The SBGT Trains utilize __(1)__ to indicate a fire in the filter bank.

If a valid fire signal is detected filter bank sprinkler flow __(2)__occur. (Assume no operator actions)

A. (1) temperature switches
   (2) will

B. (1) temperature switches
   (2) will not

C. (1) ionization detectors
   (2) will

D. (1) ionization detectors
   (2) will not

Feedback

K/A: 600000 K2.01
Knowledge of the interrelations between PLANT FIRE ON SITE and the following:
Sensors / detectors and valves

RO/SRO Rating:
2.6/2.7

Objective:
CLS-LP-41*21
Given plant conditions, predict the response of the Fire Suppression and Fire Detection Systems.

Reference:
SD-10, Standby Gas Treatment System, Section 3.2.5

Cog Level:
Low

Explanation:
There are two temperature switches to monitor the temperature of each Carbon Filter in each SBGT train.(TS 3/4) Switches VA-TS-5302-1 (VA-TS-5302-2), and VA-TS-5297-1 (VA-TS-5297-2) monitor Carbon Filter Bank No. 1 and actuate at 210°F, rising, to indicate a fire in the filter bank. Actuation of any switch will automatically open the Fire Suppression System's deluge valve for the associated carbon bank (note that the associated isolation valves must be opened manually for this system to inject)

Distractor Analysis:

Choice A: Plausible because temperature switches is correct. High temperature trips the train with inlet temp < 180°F but fire sprinkler flow will not occur until the local manual deluge vavle is opened.

Choice B: Correct Answer, see explanation

Choice C: Plausible because Ionization detectors detect the early products of combustion before they become visible smoke. High temperature trips the train with inlet temp < 180°F but fire sprinkler flow will not occur until the local manual deluge valve is opened.

Choice D: Plausible because Ionization detectors detect the early products of combustion before they become visible smoke and will not is correct due to local valve manipulations required to flow water.

65. 700000 K1.02 001

Unit Two operating at rated power with Main Generator MVAR loading at +300 MVARs.

Which one of the following completes the statement below based on these conditions?

The Main Generator component that could overheat is the __(1)__ windings.
MVARs must be lowered to less than __(2)__ IAW 2OP-27, Generator and Exciter System Operating Procedure.

A. (1) armature (stator)
   (2) +70

B. (1) field (rotor)
   (2) +70

C. (1) armature (stator)
   (2) +170

D. (1) field (rotor)
   (2) +170

Feedback

K/A: 700000 K1.02
Knowledge of the operational implications of the following concepts as they apply to GENERATOR VOLTAGE AND ELECTRIC GRID :
Over-excitation
(CFR: 41.4, 41.5, 41.7, 41.10 / 45.8)

RO/SRO Rating:
3.3/3.4

Objective:
CLS-LP-27*11f
Given plant conditions, describe the effect that a loss or malfunction of the following may have on the Main Generator: Voltage regulator (including Under and Over-Excitation)

Reference:
SD-27, Main Generator and Exciter System, Section 2.17

Cog Level:
High

Explanation:
The limitation placed on lagging MVARs (+ MVARs) of the estimated capability curve, limits operation because of excessive heating that occurs in the generator field. Since the generator is operating in an overexcited condition, a larger field current is necessary to produce the extra KVAR being supplied to the system. with the conditions given, the generator is operating outside the capabilities curve. The minimum gross MVAR requirement is 70 (positive) while the maximum gross MVAR requirement is 170 (positive).

Distractor Analysis:

Choice A: Plausible because the limitation placed on leading MVARs (- MVARs) of the estimated capability curve are less effected by Hydrogen pressure. We see that the curves come together sharply. As the system is required to supply more reactive power to the generator field, the flux in the air gap between the field and stator becomes more distorted. The distortion results in the exposed ends of the stator coils becoming overheated. As field strength is reduced, this heating accelerates. The +70 is the lower end of the operating band for the generator VARS.

Choice B: Plausible because overheating of the field is correct and the +70 is the lower end of the operating band for the generator VARS.

Choice C: Plausible because the limitation placed on leading MVARs (- MVARs) of the estimated capability curve are less effected by Hydrogen pressure. We see that the curves come together sharply. As the system is required to supply more reactive power to the generator field, the flux in the air gap between the field and stator becomes more distorted. The distortion results in the exposed ends of the stator coils becoming overheated. As field strength is reduced, this heating accelerates. The +170 is the high end of the operating band for the generator VARS.

Choice D: Correct Answer, see explanation

66. G2.01.27 001

Which one of the following is the purpose of the High Pressure Coolant Injection (HPCI) System IAW Technical Specifications Bases?

HPCI is designed to provide sufficient coolant injection to maintain the Reactor core covered during a __(1)__ Loss-Of-Coolant-Accident to maintain fuel cladding temperatures below __(2)__.

A. (1) small break
   (2) 1800°F

B.✓ (1) small break
   (2) 2200°F

C. (1) large break
   (2) 1800°F

D. (1) large break
   (2) 2200°F

Feedback

K/A: G2.01.27
Conduct of Operations
Knowledge of system purpose and/or function.
(CFR: 41.7)

RO/SRO Rating:
3.9/4.0

Objective:
CLS-LP-019*01
State the purpose of the High Pressure Coolant Injection (HPCI) System.

Reference:
U2 TS Bases
SD-19, High Pressure Coolant Injection System (HPCI), Section 1.2

Cog Level:
Low

Explanation:
The High Pressure Coolant Injection (HPCI) System was designed to provide sufficient coolant injection to maintain the Reactor core covered during a small line break Loss-Of-Coolant-Accident (LOCA) which does not result in rapid vessel depressurization, thus maintaining fuel cladding temperatures below 2200°F. The original design basis of the HPCI System was to provide part of the Emergency Core Cooling System (ECCS) function. HPCI system operation mitigated small break LOCAs where the depressurization function [Automatic Depressurization System (ADS) / SRVs] was assumed to fail.

Distractor Analysis:

Choice A: Plausible because small break is correct, and 1800°F is the number for if adequate core cooling can not be maintained by core submergence.

Choice B: Correct Answer, see explanation

Choice C: Plausible because HPCI is a high capacity, high pressure injection system which is easily mistaken for large break LOCA makeup requirements, and 1800°F is the number for if adequate core cooling can not be maintained by core submergence.

Choice D: Plausible because HPCI is a high capacity, high pressure injection system which is easily mistaken for large break LOCA makeup requirements, and 2200°F is the temperature that cladding will not exceed with core submergence.

67. G2.01.31 001

LOCA conditions exist on Unit One. CREV has failed to auto start and the CRS has ordered CREV to be manually started per the Hard Card.

Which one of the following identifies:
(1) the minimum required action(s) to start the CB Emerg Recirc Fan and
(2) the expected position indication for the CB Emerg Recirc Damper (VA-2J-D-CB) after the fan is started?

A. (1) Place the CB Emerg Recirc Fan control switch on Unit Two XU-3 panel to On
(2) Closed

B. (1) Place the CB Emerg Recirc Fan control switch on Unit Two XU-3 panel to On
(2) Open

C. (1) Simultaneously place both Units' CB Emerg Recirc Fan control switches on their respective XU-3 panel to On
(2) Closed

D. (1) Simultaneously place both Units' CB Emerg Recirc Fan control switches on their respective XU-3 panel to On
(2) Open

Feedback

K/A: G2.01.31
Conduct of Operations
Ability to locate control room switches, controls, and indications, and to determine that they correctly reflect the desired plant lineup.
(CFR: 41.10 / 45.12)

RO/SRO Rating:
4.6/4.3

Objective:
CLS-LP-37, Obj 12d
Explain the following: How to place the Control Room Ventilation system in Recirculation Mode.

Reference:
0OP-37, Control Building Ventilation System operating Procedure

Cog Level:
High

Explanation:
The controls for the CREV system are on U2 only. Indication for the CREV System is on both units. The emergency recirc damper will open when the fan is started and the open indication is red. The normal makeup damper closes on starting the fan in which the closed indication is green.
The Control Building Mechanical Equipment Room Vent Fans can only be stopped by simultaneously placing both Units' control switches in OFF.

Distractor Analysis:

Choice A: Plausible because the control switch is located on U2, but the recirc damper will open. The normal makeup damper closes.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the CB Mechanical Equipment Room Vent Fans can only be stopped by simultaneously operating both Units' control switches and the normal makeup damper closes.

Choice D: Plausible because the CB Mechanical Equipment Room Vent Fans can only be stopped by simultaneously operating both Units' control switches and the recirc damper does open.

68. G2.01.39 001

Which one of the following describes the applicable guidance of 0OI.01.02, Operations Unit Organization and Operating Practices, Attachment 1, Operations Performance Standards?

When moving a control rod __(1)__, the operator is required to stop one notch short, then single notch to the final position unless the control rod is going to position "48" (full out).

When moving a control rod __(2)__, the operator is required to perform separate single notch moves.

A. (1) 3 notches or less
   (2) 4 notches or more

B. (1) 3 notches or less
   (2) 3 notches or less

C. (1) 4 notches or more
   (2) 4 notches or more

D. (1) 4 notches or more
   (2) 3 notches or less

Feedback

K/A: G2.01.39
Conduct of Operations
Knowledge of conservative decision making practices.
(CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating:
3.6/4.3

Objective:
CLS-LP-201-D*24f
Explain the following regarding OPS-NGGC-1306 Reactivity Management Program:
The procedural requirements for positioning intermediate control rods.

Reference:
0OI-01.02, Conservative Decision Making and Reactivity Management, Attachment 1,Section 5.10

Cog Level:
Low

Explanation:
Control rod manipulations within the guidance of Attachment 1 are conservative decisions.
Per 0OI-01.02,
12) When moving a control rod four notches or more, stops one notch short, then single notches to final position unless control rod is going to position "48" (full out).
13) When moving a control rod three notch or less, performs separate single notch moves.

Distractor Analysis:

Choice A: Plausible because since 3 notches provides for sufficient rod movement to allow for stopping conservatively one notch short and moving control rods 4 notches or more by single notch is conservative.

Choice B: Plausible because since 3 notches provides for sufficient rod movement to allow for stopping conservatively one notch short and 3 notches or less is correct.

Choice C: Plausible because 4 notches or more is correct and moving control rods 4 notches or more by single notch is conservative.

Choice D: Correct Answer, see explanation

69. G2.02.04 001

Which one of the following identifies where the Electric Fire Pump can be started from?

The Electric Fire Pump can be manually started locally:

A. only.

B. and at the Unit One RTGB only.

C. and at the Unit Two RTGB only.

D. and at either Unit One or Unit Two RTGBs.

Feedback

K/A: G2.02.04
EQUIPMENT CONTROL
(multi-unit license) **Ability to explain the variations in control board/control room layouts, systems, instrumentation, and procedural actions between units at a facility.**
(CFR: 41.6 / 41.7 / 41.10 / 45.1 / 45.13)

RO/SRO Rating:
3.6/3.6

Objective:
None

Reference:
0OP-41, Fire Suppression Systems Operating Procedure, Section 8.3
SD-41, Fire Suppression Systems, Section 3.2.2

Cog Level:
Low

Explanation:
Ability to explain the variation is identifing the difference in location of the controls.

The Electric Fire Pump can be started
1. Automatically (Low system pressure)
2. Manually from the Control Room (Panel XU-69 which is only located on Unit One)
3. Manually from local control panel.

Distractor Analysis:

Choice A: Plausible because the majority of License Operator simulator training is performed on U2 simulator which does not have Panel XU-69.

Choice B: Correct Answer, see explanation

Choice C: Plausible because controls would exist on Unit Two however they do not. This is a difference between Units

Choice D: Plausible because controls would exist on Unit Two however they do not. This is a difference between Units

70. G2.02.25 001

Which one of the following describes the bases for the Minimum Critical Power Ratio (MCPR) Safety Limit IAW Technical Specifications Bases 2.1.1, Reactor Core Safety Limits?

The MCPR Safety Limit ensures that:

A. the calculated total oxidation shall no where exceed 0.17 times the total cladding thickness before oxidation.

B. the calculated changes in core geometry shall be such that the core remains amenable to cooling.

C. cladding plastic strain remains less than 1%.

D. during normal operation and during Anticipated Operational Occurrences, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Feedback

K/A: G2.02.25
EQUIPMENT CONTROL
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
(CFR: 41.5 / 41.7 / 43.2)

RO/SRO Rating:
3.2/4.2

Objective:
CLS-LP-200-B*03
State each TS Safety Limit and discuss the basis for each of the Safety Limits.

Reference:
U2 TS Bases

Cog Level:
Low

Explanation:
Requires knowledge of TS Safety Limit Bases and the ability to distinguish between Safety Limits and Operating Limits. The MCPR fuel cladding integrity SL ensures that during normal operation and during AOOs, at least 99.9% of the fuel rods in the core do not experience transition boiling.

Distractor Analysis:

Choice A: Plausible because since this is ECCS acceptance criteria.

Choice B: Plausible because since this is ECCS acceptance criteria.

Choice C: Plausible because since this is the basis for the LHGR limit.

Choice D: Correct Answer, see explanation

71. G2.02.39 001

Unit Two startup is in progress when the operating CRD Pump trips.

The following plant conditions exist:

CRD pumps	Unavailable
Reactor Pressure	850 psig
Charging header pressure	875 psig
CRD ACCUM LO PRESS/HI LEVEL	In alarm
Control Rod 18-19 Full core display	Amber Accumulator light is lit
Control Rod 18-19 Full core display	Red Full Out light is lit

Which one of the following describes the required action IAW Technical Specifications 3.1.5, Control Rod Scram Accumulators and 0AOP-02.0, Control Rod Malfunction/Misposition?

A. A manual reactor scram is required Immediately.

B. Restore charging water header to >950 psig within 20 minutes or insert a manual reactor scram.

C. Fully insert control rod 18-19 immediately and declare INOP within 1 hour.

D. A manual reactor scram is required <u>only</u> if a second accumulator alarm is received.

Feedback

K/A: G2.02.39
EQUIPMENT CONTROL
Knowledge of less than or equal to one hour Technical Specification action statements for systems.
(CFR: 41.7 / 41.10 / 43.2 / 45.13)

RO/SRO Rating:
3.9/4.5

Objective:
CLS-LP-008-B*10
Given plant conditions, determine proper operator actions if no CRD pumps are operating.

Reference:
Unit 2 Tech Spec 3.1.5, Control Rod Scram Accumulators, Condition D

Cog Level:
High

Explanation:
Immediate scram required by Tech Spec 3.1.5, Conditions C & D.
The reactor must be immediately scrammed if either the Required Action and associated Completion Time associated with loss of the CRD charging pump (Required Actions B.1 and C.1) cannot be met. This ensures that all insertable control rods are inserted and that the reactor is in a condition that does not require the active function (i.e., scram) of the control rods.
Scram also required by supplemental actions of 0AOP-02.0.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because with reactor pressure $\geq$ 950 psig combined with charging header pressure < 940 psig, 20 minutes is allowed for restoration of charging water header pressure.

Choice C: Plausible because accumulator alarm could be due to High water level which would still provide for sufficient accumulator pressure to fully insert the control rod. Revision 12 of 2AOP-02.0 provided guidance on time frame to IMMEDIATELY insert a manual scram upon receipt of the first HCU low pressure alarm (A-07 6-1, confirmed by amber light on Full Core Display).

Choice D: Plausible because waiting for the second accumulator is applicable to reactor pressures $\geq$ 950 psig for restoration of charging water header pressure.

72. G2.03.14 002

Unit One is operating at 50% power with the HWC System operating in Auto.

Which one of the following choices completes the statements below?

The reactor power input to HWC System is determined by total __(1)__ flow.
If reactor power is raised, the Turbine Building radiation levels will increase due to the increased production of __(2)__.

A. (1) Steam
(2) Ammonia (NH_3)

B. (1) Feedwater
(2) Ammonia (NH_3)

C. (1) Steam
(2) Nitrates (NO_3)

D. (1) Feedwater
(2) Nitrates (NO_3)

Feedback

K/A: G2.03.14
Radiation Control
Knowledge of radiation or contamination hazards that may arise during normal, abnormal, or emergency conditions or activities.
(CFR: 41.12 / 43.4 / 45.10)

RO/SRO Rating: 3.4/3.8

Objective: CLS-LP-59*14
14. Explain why background radiation levels outside primary containment increase when the HWC System is placed in service.
15. State the parameter used for the reactor power level reference input to the hydrogen injection flow controller, and explain why it is used.

Reference:
SD-59, Revision 14, Page 8, Section 1.3.2

Cog Level: Low

Explanation:
The implementation of Hydrogen Water Chemistry (H2 injection) alters the Nitrogen-16 carryover ratio. The net production of Nitrogen-16 is not influenced by Hydrogen injection. The excess Hydrogen injected into the reactor coolant creates the driving force to shift the Nitrogen-16 distribution ratio, resulting in a larger fraction of the Nitrogen-16 forming volatile Ammonia and a smaller fraction forming Nitrites and Nitrates. This additional volatile Ammonia is then carried over in the reactor steam resulting in higher background radiation levels. Any increase in Hydrogen injection rates will result in a proportional increase in background radiation levels and vice-versa. DFCS provides power input signals to systems (RWM, HWC, Recirc) via total steam flow and total feedwater flow.
HWC utilizes Feedwater flow from DFCS for power input

Distractor Analysis:
Choice A: Plausible because DFCS steam flow does provide power input to the RWM and is often confused with feedwater flow and increased production of Ammonia (NH_3) is correct.

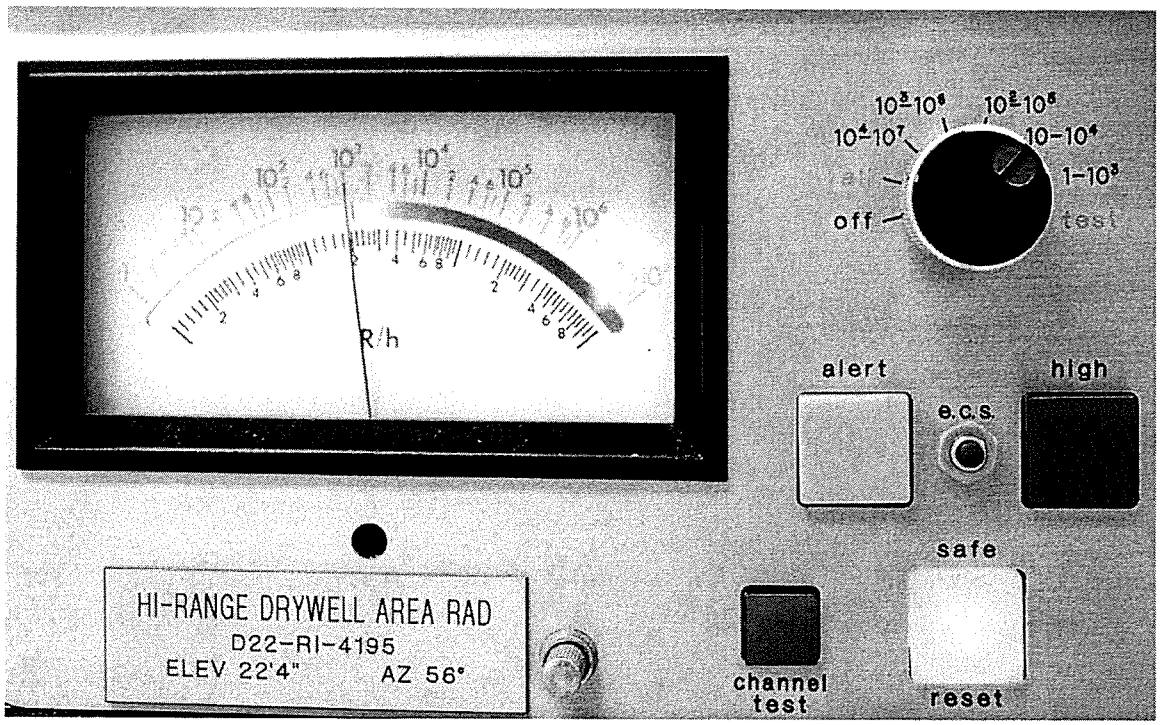
Choice B: Correct Answer

Choice C: Plausible because DFCS steam flow does provide power input to the RWM and is often confused with feedwater flow and raising power with HWC in AUTO raises H2 injection rate which raises Ammonia (NH_3) carryover. Increased Nitrates (NO_3) are removed by RWCU and do not cause TB rad levels to rise.

Choice D: Plausible because feedwater is correct and raising power with HWC in AUTO raises H2 injection rate which raises Ammonia (NH_3) carryover. Increased Nitrates (NO_3) are removed by RWCU and do not cause TB rad levels to rise.

SRO Only Basis: N/A

73. G2.03.15 001



Which one of the following identifies the current Drywell radiation level?

A. ~20 R/h

B. ~200 R/h

C. ~1000 R/h

D. ~10000 R/h

Feedback

K/A: G2.03.15
Radiation Control
Knowledge of radiation monitoring systems, such as fixed radiation monitors, portable survey instruments, personnel monitoring equipment, etc.
(CFR: 41.12 / 43.4 / 45.9)

RO/SRO Rating:
2.9/3.1

Objective:
CLS-LP-11.1*03a
Describe the function/operation of the following: Drywell High Range Radiation Monitors

Reference:
SD-11.1, Area Radiation Monitoring System, Section 2.5

Cog Level:
Low

Explanation:
Drywell high range area monitors provide indications of gross fuel failure and are used to determine emergency plan emergency action level associated with abnormal core conditions. With the function switch in the E1-E4, meter readings are taken from the lower scale between 10 - 10000 R/h. Current indication of 200 R/h

Distractor Analysis:

Choice A: Plausible if function switch is not taken into account would be 20 R/h.

Choice B: Correct Answer, see explanation

Choice C: Plausible if read directly off the upper scale

Choice D: Plausible if read off the upper scale and adjusted by a factor of 10 for function switch position.

74. G2.04.01 002

Which one of the following radiation annunciators requires entry into RRCP?

A. *RBCCW LIQUID PROCESS RAD HIGH*

B. *RX BLDG ROOF VENT RAD HIGH*

C. *PROCESS RX BLDG VENT RAD HIGH*

D. *PROCESS RX BLDG VENT RAD HI-HI*

Feedback

K/A: G2.04.01
Emergency Procedures / Plan
Knowledge of EOP entry conditions and immediate action steps.
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating: 4.6/4.8

Objective: LOI-CLS-LP-300-N*002
2. Given plant conditions, determine if 0EOP-04-RRCP should be entered.

Reference:
RRCP

Cog Level: Low

Explanation: *Brunswick does not have any immediate operator actions in any EOP.*
Annunciator requires immediate operator action of entry into RRCP. RRCP provides guidance to the operator for minimizing off-site radioactivity releases up to and including events involving substantial degradation of all of the fission product barriers (e.g., fuel, fuel clad, reactor vessel pressure boundary, primary containment, and secondary containment).
RRCP and SCCP are used concurrently to control releases from primary systems. This procedure controls non-primary system releases through actions incorporated in the non-PSTG legs of the procedure.

Distractor Analysis:
Choice A: Plausible because SCCP entry would be appropriate.

Choice B: Correct Answer

Choice C: Plausible because is easily confused with the roof vent alarm and is a SCCP entry condition.

Choice D: Plausible because this annunciator provides indication of Secondary Containment auto isolation setpoint and is easily confused with the roof vent alarm.

SRO Only Basis: N/A

75. G2.04.09 001

An ATWS has occurred on Unit One with the following plant conditions:

Reactor Water Level	130 inches (stable)
Injection Systems	CRD
Reactor Power	APRM downscale lights are illuminated
Control Rods	19 rods failed to insert
SRVs	All closed
Reactor Pressure	920 psig and stable
Suppression Pool Temp.	92° F

Which one of the following choices completes the statement below IAW LPC?

Reactor Recirculation pumps __(1)__ required to be tripped and the SLC Pumps __(2)__ required to be started.

A. (1) are not
(2) are not

B. (1) are not
(2) are

C. (1) are
(2) are not

D. (1) are
(2) are

Feedback

K/A: G2.04.09
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.
(CFR: 41.10 / 43.5 / 45.13)
e.g. is an abbreviation for the latin phrase exempli gratia. When you mean "that is" use i.e. Since e.g. indicates a partial list, it would be redundant to add etc. at the end of a list introduced by this abbreviation. (from answerbag.com) The recirc pumps and SLC pumps are used for mitigation of the ATWS.

RO/SRO Rating:
3.8/4.2

Objective:
CLS-LP-300-E*017
Compare and contrast the operator actions for emergency depressurization with an ATWS condition present versus those with all control rods inserted.

Reference:
0OI-37.5

Cog Level:
High

Explanation:
Reactor recirc pumps are evaluated during the ATWS and are tripped to reduce core flow thereby reducing reactor power. If power is less than 2% then tripping the pumps is not required. SLC is injected into the reactor to provide an alternate means of shutting down the reactor. It is required to be injected prior to exceeding 110° F in the torus. With SRVs closed and temperature at 92 with no additional heat load to the torus SLC injection is not required.

Distractor Analysis:

Choice A: Correct Answer, see explanation.

Choice B: Plausible because recirc pumps are not required to be shutdown is correct and if the torus was in jeopardy of reaching 110 then this would be correct.

Choice C: Plausible because if power was greater than 2% the recirc pumps are required to be shutdown is correct and the torus is not in jeopardy of reaching 110 so this is correct.

Choice D: Plausible because if power was above 2% and the torus temperature in danger of reaching 110 then these actions would be correct.

76. S204000 G2.02.25 003

With Unit Two operating at power which one of the following choices completes the statements below IAW Technical Specification 3.6.1.3, Primary Containment Isolation Valves (PCIVs)?

If the G31-F001, RWCU Inlet Inboard Isolation Valve is inoperable, the line is required to be isolated in __(1)__ and verify the affected penetration flow path is isolated once per 31 days.

This 31 day verification of the G31-F004, RWCU Inlet Outboard Isolation Valve, breaker and valve position __(2)__.

A. (1) 2 hours
(2) may be accomplished remotely by Administrative means

B. (1) 8 hours
(2) may be accomplished remotely by Administrative means

C. (1) 2 hours
(2) must be accomplished by actual "hands on" verification at the components

D. (1) 8 hours
(2) must be accomplished by actual "hands on" verification at the components

Feedback

K/A: 204000 G2.02.25
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
Reactor Water Cleanup System
(CFR: 41.5 / 41.7 / 43.2)
There are no safety limits associated with RWCU system, so question is written directly to the TS.

RO/SRO Rating:
3.2/4.2

Objective:
CLS-LP-14, Obj. 13
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR, determine the required action(s) to be taken in accordance with Technical Specifications associated with the RWCU system.

Reference:
Tech Spec 3.6.1.3 Primary Containment Isolation Valves (PCIVs)

Cog Level:
Low

Explanation:
Tech Spec 3.6.1.3 PCIVs Condition A applies; Required Action A.1 must be completed within 8 hours; Required Action A.2 must be completed once per 31 days. Bases 3.6.1.3 allows completion of A.2 by administrative means for the given conditions.

Distractor Analysis:

Choice A: Plausible because 2 hours is the required completion time for Required Action B.2

Choice B: Correct Answer, see explanation

Choice C: Plausible because 2 hours is the required completion time for Required Action B.2; plausible because "hands on" verification is required in areas other than high radiation areas

Choice D: Plausible because "hands on" verification is required in areas other than high radiation areas

SRO Basis:
10 CFR 55.43(b)-2, Facility operating limitations in the TS and their bases.
Knowledge of TS bases that is required to analyze TS required actions and terminology.

77. S212000 A2.12 002

Unit One is operating at full power when the following plant conditions occur:

- Part time load shedding has been implemented
- Load Reject Signal received
- Line 31 (Whiteville Line) PCBs red lights are lit
- Line 31 (Whiteville Line) white VOLT lights are not lit
- All other line PCBs green lights are lit
- *230 KV BUS 1A BUS POT UNDERVOLTAGE* is in alarm
- *230 KV BUS 1B BUS POT UNDERVOLTAGE* is in alarm

Which one of the following identifies the initial RPS trip signal and the procedure which contains the guidance to trip the Whiteville Line PCBs?

A. Turbine Control Valve Fast Closure;
0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses.

B. Turbine Stop Valve Closure;
0AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses.

C. Turbine Control Valve Fast Closure;
0AOP-22, Grid Instability.

D. Turbine Stop Valve Closure;
0AOP-22, Grid Instability.

Feedback

K/A: 212000 A2.12
Ability to (a) predict the impacts of the following on the REACTOR PROTECTION SYSTEM ; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal conditions or operations:
Main turbine stop control valve closure
(CFR: 41.5 / 45.6)

RO/SRO Rating:
4.0/4.1

Objective:
CLS-LP-03, Obj. 8.
List the RPS trip signals, including setpoints and how/when each signal is bypassed.

Reference:
SD-03, Reactor Protection System
AOP-36.1, Loss of Any 4160V Buses or 480V E-Buses

Cog Level:
High

Explanation:
A load reject signal at any reactor power level will cause a turbine control valve fast closure scram. The load reject signal does not input into the turbine stop valve closure scram logic. During a loss of offsite power, if the grid is lost all PCBs are opened per 0AOP-36.1.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Incorrect Load reject initiates a TCV fast closure scram not a TSV. A misconception of the difference between TCV and TSV scrams may cause a student to select this answer.

Choice C: Incorrect. 0AOP-22 does not have an action for loss of grid only for degraded conditions.

Choice D: Incorrect. Load reject initiates a TCV fast closure scram not a TSV. A misconception of the difference between TCV and TSV scrams may cause a student to select this answer. 0AOP-22 does not have an action for loss of grid only for degraded.

SRO Basis:
10 CFR 55.43(b)-5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

78. S214000 A2.03 001

The Unit is at 30% power during reactor startup.
The operator withdraws control rod 26-27 to position 48.
The following indications are noted:

- *ROD DRIFT* alarm seals in
- *ROD OVER TRAVEL* alarm seals in
- Rod 26-27 full core display red light out

Which one of the following identifies:
(1) the indication that would be displayed on the four-rod group display and
(2) the minimum required action(s) for the inoperable control rod IAW Technical Specifications?

A. (1) 48
(2) Fully insert control rod 26-27 and disarm the HCU

B. (1) 48
(2) Fully insert control rod 26-27 and disarm the HCU;
verify ≥12 control rods are withdrawn and implement GP-11, Second Operator Rod Sequence Checkoff Sheets

C. (1) Blank
(2) Fully insert control rod 26-27 and disarm the HCU

D. (1) Blank
(2) Fully insert control rod 26-27, disarm the HCU;
verify ≥12 control rods are withdrawn, and implement GP-11, Second Operator Rod Sequence Checkoff Sheets

Feedback

K/A: 214000 A2.03
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:
Overtravel/in-out
(CFR: 41.5 / 45.6)

RO/SRO Rating:
3.6/3.9

Objective:
CLS-LP-07 Obj 5b
Given plant conditions, determine if the following conditions exist: Indications of an uncoupled control rod.

Reference:
SD-07, Reactor Manual Control System
Tech Spec 3.1.3, Control Rod Operability
GP-11, Second Operator Rod Sequence Checkoff Sheets

Cog Level:
Low

Explanation:
If the control rod is in the overtravel out position, the corresponding digital indicator will be blank since the magnet will not be near any of the 00 to 48 reed switches. IAW TS the rod is declared inoperable then inserted to 00 (within 3 hours) and disarmed (within 4 hours). TS 3.1.6 if the RWM is inoperable then if $\geq$12 control rods are withdrawn GP-11 would be implemented, unless the rod is at 00 and is not intended to be moved.

Distractor Analysis:

Choice A: Plausible because the full in and 00 indications are at the same point or the examinee may think that the rod may settle to the 48 position.

Choice B: Plausible because the full in and 00 indications are at the same point or the examinee may think that the rod may settle to the 48 position. These are TS actions for an inoperable RWM, not control rod.

Choice C: Correct Answer, see explanation.

Choice D: Plausible because this is the correct indication but these are TS actions for an inoperable RWM, not control rod.

SRO Basis:
10 CFR 55.43(b)-2, Facility operating limitations in the technical specifications and their bases. Application of required actions statements.

79. S217000 G2.04.08 002

Given the following ATWS conditions on Unit Two:

2A CRD Pump	Overcurrent trip
2B CRD Pump	Shaft uncoupled
HPCI System	Under Clearance
SLC	Both squib valves failed to fire
RCIC	Running for LEP-03, Alternate Boron Injection, with an unisolable steam supply leak
SRHR Equip Room	180°F
Suppression Pool Level	-24 inches
Reactor Power	10%
Reactor Water Level	160 inches

Which one of the following completes the statement below based on the conditions above?

The RCIC system must __(1)__ which is required by __(2)__.

A. (1) be isolated
(2) SCCP

B. (1) be isolated
(2) 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity

C✓ (1) remain running
(2) SCCP

D. (1) remain running
(2) 0AOP-05.0, Radioactive Spills, High Radiation, and Airborne Activity

Feedback

K/A: 217000 G2.04.08
Knowledge of how abnormal operating procedures are used in conjunction with EOPs.
Reactor Core Isolation Cooling System (RCIC)
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.8/4.5

Objective:
CLS-LP-300-B, Obj.19
Given a situation where a conflict exists between AOP or ASSD requirements and EOP guidance, determine the correct operator action.

Reference:
AOP-05, Radioactive Spills, High Radiation and Airborne Activity
SCCP

Cog Level:
High

Explanation:
EOP action that supercedes the AOP action is what the question is asking.
AOP-5 does have a step to isolate the system that is leaking, but SCCP overrides that if the system is required by EOPs. With the ATWS the RCIC system is required for alternate boron injection therefore should remain running per the direction from SCCP.

Distractor Analysis:

Choice A: Plausible because SCCP/AOP-5 does have a step to isolate the system that is leaking and reasonable for the student to think this direction comes from SCCP.

Choice B: Plausible because SCCP/AOP-5 does have a step to isolate the system that is leaking

Choice C: Correct Answer, see explanation

Choice D: Plausible for student to think the direction to remain running comes from AOP-5.0.

SRO Basis:
10 CFR 55.43(b)-5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

80. S233000 G2.04.06 001

Unit Two was operating at rated power with the following conditions:

- A dual Unit Loss Of Offsite Power (LOOP)
- Spent Fuel Pool level is lowering rapidly
- RRCP has been entered due to high rad conditions on the refuel floor

Which one of the following is the first makeup source to be used for filling the fuel pool and identifies the procedure to perform the action?

A. Emergency Diesel Makeup Pump IAW 0AOP-38.0, Loss of Fuel Pool Cooling

B. ✓ RHR B Loop IAW 0AOP-38.0, Loss of Fuel Pool Cooling

C. Emergency Diesel Makeup Pump IAW 0EDMG-002, Spent Fuel Pool Makeup/Spray and Refuel Floor Enhanced Ventilation under Conditions of Extreme Damage

D. RHR B Loop IAW 0EDMG-002, Spent Fuel Pool Makeup/Spray and Refuel Floor Enhanced Ventilation under Conditions of Extreme Damage

Feedback

K/A: 233000 G2.04.06
Knowledge of EOP mitigation strategies.
Fuel Pool Cooling and Clean-up
(CFR: 41.10 / 43.5 / 45.13)
There are no direct EOP actions associated with FPC, a loss of level in the fuel pool will cause entry into RRCP which is an EOP. So these actions are mitigation strategies to RRCP.

RO/SRO Rating:
3.7/4.7

Objective:
CLS-LP-13, Obj.11
State the sources of makeup water for the Fuel Pool in order of preference.

Reference:
AOP-38.0, Loss of Fuel Pool Cooling
EDMG-02, Spent Fuel Pool Makeup/Spray and Refuel Floor Enhanced Ventilation under Conditions of Extreme Damage

Cog Level:
High

Explanation:
The order of the makeup sources is from the normal fill, Demin water hose stations, Fire protection hose stations, demin water through RHR keepfill, and then other sources that are not service water. For a high capacity water source and the gates installed RHR Loop B would be used via the FPC system. With a LOOP the demin pumps have no power. If no other sources are available then the procedure has injection from the EDMP.

Distractor Analysis:

Choice A: Plausible because although this is a makeup source it is not the preferred source (last resort per the procedure) and is performed per the EDMG procedures. Although upon entering the AOP there is a step to start lining this system up for injection because of the time required to get all of the hoses run in the procedure up to the fuel pool.

Choice B: Correct Answer, see explanation

Choice C: Plausible because although this is a makeup source it is not the preferred source (last resort per the procedure). Although upon entering the AOP there is a step to start lining this system up for injection because of the time required to get all of the hoses run in the procedure up to the fuel pool.

Choice D: Plausible because RHR is the high capacity source that will need to be used, but the EDMG procedure does not provide this guidance.

SRO Basis:
10 CFR 55.43(b)-5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

81. S239002 G2.02.25 002

With Unit Two at rated power, which one of the following identifies:

(1) the required number of SRVs with safety function operable IAW Technical Specification LCO 3.4.3, Safety/Relief Valves and
(2) the overpressurization event that bounds this number of required operable SRVs IAW Technical Specification Bases 3.4.3, Safety/Relief Valves?

A. (1) 9
(2) an ATWS with MSIV closure

B. (1) 10
(2) an ATWS with MSIV closure

C. (1) 9
(2) an MSIV closure with reactor scram

D. (1) 10
(2) an MSIV closure with reactor scram

Feedback

K/A: 239002 G2.02.25
Knowledge of the bases in Technical Specifications for limiting conditions for operations and safety limits.
Safety Relief Valves
(CFR: 41.5 / 41.7 / 43.2)

RO/SRO Rating:
3.2/4.2

Objective:
CLS-LP-25, Obj. 10
Given plant conditions and TS, including the Bases, TRM, ODCM, and COLR determine the required actions to be taken in accordance with TS associated with the Reactor Recirculation System. (SRO only)

Reference:
TS 3.4.3 Safety/Relief Valves (SRVs) and bases document

Cog Level:
Low

Explanation:
TS 3.4.3 states 10 must be operational for the safety function, the bases states the reason, ATWS.

Distractor Analysis:

Choice A: Plausible because the bases states that 9 are required for the MSIV closure.

Choice B: Correct Answer, see explanation

Choice C: Plausible because the bases states that 9 are required for the MSIV closure and the MSIV closure is not the binding failure mode.

Choice D: Plausible because 10 are required for the ATWS and the MSIV closure is not the binding failure mode.

SRO Basis:
10 CFR 55.43(b)-2, Facility operating limitations in the technical specifications and their bases.
This is knowledge of tech spec bases to determine the reason 10 are required.

82. S262002 G2.01.23 001

Both Units are operating at full power when the Main Stack Rad Monitor lost its normal power supply.

Which one of the following identifies the procedure that contains the steps to transfer the Main Stack Rad Monitor to its alternate power supply?

A. 1OP-52, 120 Volt AC UPS, Emergency, and Conventional Electrical Systems Operating Procedure

B.✓ 2OP-52, 120 Volt AC UPS, Emergency, and Conventional Electrical Systems Operating Procedure

C. 1APP UA-03 6-3, *PROCESS SMPL OG VENT PIPE DNSC/INOP*

D. 2APP UA-03 6-3, *PROCESS SMPL OG VENT PIPE DNSC/INOP*

Feedback

K/A: 262002 G2.01.23
Ability to perform specific system and integrated plant procedures during all modes of plant operation.
Uninterruptable Power Supply (A.C./D.C.)
(CFR: 41.10 / 43.5 / 45.2 / 45.6)

RO/SRO Rating:
4.3/4.2

Objective:
CLS-LP-11.0, 15a
Given plant conditions and a trip or failure of one of the following Radiation Monitors, determine appropriate plant response and use procedures to determine the actions required to control and/or mitigate the consequences of the event: a. Main Stack.

Reference:
2OP-52, Section 8.7 Stack Radiation Monitor UPS Power Supply Transfer

Cog Level:
Low

Explanation:
The normal power supply for the Main Stack Rad Monitor is from Unit Two. On a loss of power from the normal power supply the operators will need to transfer to the alternate power supply. This direction is only in the U2 procedure. There is no directions to perform this in the U1 procedure or the APPs for either Unit.

Distractor Analysis:

Choice A: Plausible if the student wrongly believes the normal power supply is from Unit 1 rather than Unit 2.

Choice B: Correct Answer, see explanation.

Choice C: Plausible because the downscale / inop annunciator will be actuated on a loss of power but the APPs do not address transfer of power to backup supply.

Choice D: Plausible because the downscale / inop annunciator will be actuated on a loss of power but the APPs do not address transfer of power to backup supply. U2 is the normal power supply to the rad monitor.

SRO Basis:
10 CFR 55.43(b)-5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

83. S295006G 2.02.22 002

Following a reactor scram on Unit Two the following conditions exist:

0900 RHR Pump 2B tripped on overcurrent while placing Suppression Pool Cooling in service.

1100 The RSP is exited to 0GP-05, Unit Shutdown, while in Mode 3.

Which one of the following completes the statement below?

The Technical Specification 3.5.1, ECCS - Operating, LCO Action Statement is required to be entered at __(1)__ IAW 0EOP-01-UG, User's Guide.

RHR Loop B Operability is required to be restored within __(2)__ IAW Tech Spec 3.5.1, ECCS - Operating.

A. (1) 0900
(2) 7 days

B. (1) 1100
(2) 7 days

C. (1) 0900
(2) 14 days

D. (1) 1100
(2) 14 days

Feedback

K/A: S295006G 2.02.22
Knowledge of limiting conditions for operations and safety limits.
SCRAM
(CFR: 41.5 / 43.2 / 45.2)

RO/SRO Rating:
4.0/4.7

Objective:
CLS-LP-300-C*11
Given plant conditions, the Unit Shutdown Procedure (GP-05), and the Reactor Scram Procedure, determine if conditions allow exiting the Reactor Scram Procedure.

Reference:
0EOP-01-UG, Revision 55, Page 31, Section 3.5
Tech Spec 3.5.1, ECCS Operating

Cog Level:
High

Explanation:
The EOPs authorize actions outside of technical specifications to mitigate the consequences of an emergency condition. The EOPs also provide for returning the system or component to service. If the system or component is not returned to its standby or operable condition prior to exiting the EOPs, then the appropriate limiting condition of operation shall be implemented in accordance with Technical Specifications. The starting time for the limiting condition of operation is the time that the EOPs were exited.
In order to exit EOP, compatibility with GP-05 along with active LCOs need to be implemented. RHR Pump 2B tripping on overcurrent requires declaring RHR Loop B Inoperable for LPCI (Modes 1,2 &3), TS 3.5.1 Condition A (A1)requires Restore low pressure ECCS injection/spray subsystem to OPERABLE status within 7 days.

Distractor Analysis:

Choice A: Plausible because LCO start time is wrong (time occurred while executing EOPs) and 7 days is correct.

Choice B: Correct Answer, see explanation

Choice C: Plausible because LCO start time is wrong (time occurred while executing EOPs) and 14 days is wrong completion time (would be correct for HPCI within the same TS)

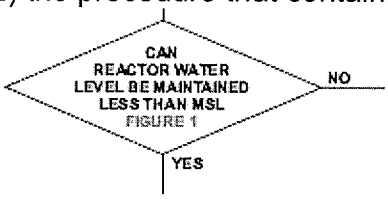
Choice D: Plausible because LCO start time is correct and 14 days is wrong completion time (would be correct for HPCI within the same TS)

SRO Basis:
Requires knowledge of EOP bases and greater than 1 hour TS completion times

84. S295008 A2.05 002

Following a scram on Unit Two, which one of the following identifies:

(1) the response of reactor water level when an SRV is opened and
(2) the procedure that contains the following decision step?



A. (1) Shrink
(2) RSP

B. (1) Shrink
(2) RVCP

C✓ (1) Swell
(2) RSP

D. (1) Swell
(2) RVCP

Feedback

K/A: 295008 A2.05
Ability to determine and/or interpret the following as they apply to HIGH REACTOR WATER LEVEL:
Swell
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
2.9/3.1

Objective:
CLS-LP-300-C, Obj.10
Given plant conditions and the RSP, determine the required operator actions.

Reference:
RSP
RVCP
0OI-37.3, Reactor Scram Procedure Basis Document
0OI-37.4, Reactor Vessel Control Procedure Basis Document

Cog Level:
High

Explanation:
Opening of the SRV will cause the reactor water level to swell up due to the reduction in pressure in the vessel. The given decision block is from the RSP.

Distractor Analysis:

Choice A: Plausible if the examinee thinks that opening the SRV would reduce the water volume in the RPV.

Choice B: Plausible if the examinee thinks that opening the SRV would reduce the water volume in the RPV and that the given decision step is applicable to the level leg of RVCP.

Choice C: Correct Answer, see explanation

Choice D: Plausible because examinee may think that the given decision step is applicable to the level leg of RVCP.

SRO Basis:
10 CFR 55.43(b)-5, Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

85. S295014 G2.01.25 001

Unit Two is operating at 74% power with the Feedwater Temperature at 405°F.

The FW-V120, FW HTRS 4 & 5 BYP VLV, is inadvertently opened by mechanics. The valve is bound in mid position and cannot be reclosed.

Conditions are now stable with reactor power at 81% and Feedwater Temperature at 306°F.

(Reference provided)

Which one of the following identifies whether continued operation is allowed including the required actions or procedures?

Continued operation:

A. is not allowed
and a reactor shutdown per 0GP-05, Unit Shutdown, is required IAW 2OP-32, Condensate and Feedwater System Operating Procedure.

B. is not allowed
and a reactor shutdown per 0GP-05, Unit Shutdown, is required IAW TS 3.1.2, Reactivity Anomalies.

C. is allowed
provided that 0GP-13, Increasing Unit Capacity At End Of Core Cycle, is implemented.

D. is allowed
provided that the FWTR Power to Flow Map is implemented.

Feedback

K/A: 295014 G2.01.25
Ability to interpret reference materials, such as graphs, curves, tables, etc.
Inadvertent Reactivity Addition
(CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating:
3.9/4.2

Objective:
CLS-LP-34, Obj.11c
Given plant conditions, describe the effect a loss/malfunction of the feedwater heaters will have on:
Feedwater Temperature

Reference:
2OP-32, Attachment 4 (provided)

Cog Level:
High

Explanation:
From Attachment 4 of 2OP-32 operation is outside of the allowable range (<110.3°F) therefore not allowed and will require a Unit shutdown IAW GP-05 per the direction of 2OP-32.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the examinee may believe that the direction to shutdown is driven from Tech Spec 3.1.2, Reactor Anomalies

Choice C: Plausible because the examinee could wrongly apply the direction of GP-13.

Choice D: Plausible because the examinee could believe that the use of the FWTR Power to Flow Map is applicable for the given situation.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (43(b)(5) a. Assessement of the plant conditions and then prescribing the shutdown IAW the GP.

86. S295021 A2.03 002

While in Mode 4 a loss of Shutdown Cooling (SDC) occurs.

Which one of the following completes both statements?

The minimum required Reactor Water Level to support Natural Circulation is __(1)__ inches.

An Alert declaration is first required after an unplanned RPV pressure increase greater than __(2)__ psig due to a loss of RCS cooling.

A. (1) 200
   (2) 135

B.✓ (1) 200
   (2) 10

C. (1) 254
   (2) 135

D. (1) 254
   (2) 10

Feedback

K/A: S295021 A2.03
Ability to determine and/or interpret the following as they apply to LOSS OF SHUTDOWN COOLING :
Reactor water level
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.5/3.5

Objective:
CLS-LP-120*06
Describe how to determine when natural circulation exists within the Reactor Vessel.

Reference:
0AOP-15, Loss of Shutdown Cooling
0PEP-2.1, EAL Flowcharts

Cog Level:
High

Explanation:
During conditions in which there is no circulation, the reactor vessel water level, as read on *B21-LI-R605A(B),* should be maintained between 200" and 220", or as directed by the Shift Superintendent based on plant conditions, until forced circulation is restored.
IAW PEP-2.1, Section CA3 "Inability to Maintain Plant in Cold Shutdown" an "unplanned reactor pressure increase of >10 psig due to a loss of SDC" requires an ALERT declaration.

Distractor Analysis:

Choice A: Plausible because 135 psig reactor pressure is a setpoint associated with SDC that could be confused with the correct answer.

Choice B: Correct Answer, see explanation

Choice C: Plausible because 254 inches is the level of the MSLs and could be confused with Natural Circulation level due to the requirement to be at this level during alternate SDC.
Plausible because 135 psig reactor pressure is a setpoint associated with SDC that could be confused with the correct answer.

Choice D: Plausible because 254 inches is the level of the MSLs and could be confused with Natural Circulation level due to the requirement to be at this level during alternate SDC

SRO Only Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (43(b)(5)a.
Requires assessing plant conditions (LOOP, Mode 3, power availability, impact of power losses) and prescribing correct section of a procedure to provide DHR.

87. S295023G 2.04.49 001

While performing refueling activities on Unit Two, a spent fuel bundle was dropped and the following alarms were received:

AREA RAD REFUEL FLOOR HIGH
PROCESS RX BLDG VENT RAD HIGH

Which one of the following identifies:

(1) the required immediate operator action and
(2) the bases for the performance of this action IAW 0AOP-05.0 Radioactive Spills, High Radiation and Airborne Activity, and Technical Specifications Bases?

A. (1) Manually start Standby Gas Treatment (SBGT)
(2) Ensures control room operators will not exceed 2 Rem TEDE

B. (1) Manually start Standby Gas Treatment (SBGT)
(2) Ensures control room operators will not exceed 5 Rem TEDE

C. (1) Manually start Control Room Emergency Ventilation (CREV)
(2) Ensures control room operators will not exceed 2 Rem TEDE

D. (1) Manually start Control Room Emergency Ventilation (CREV)
(2) Ensures control room operators will not exceed 5 Rem TEDE

Feedback

K/A: S295023G 2.04.49
Ability to perform without reference to procedures those actions that require immediate operation of system components and controls.
Refueling Accidents
(CFR: 41.10 / 43.2 / 45.6)

RO/SRO Rating:
4.6/4.4

Objective:
CLS-LP-302-J*02
Given plant conditions with spent fuel damage and a high airborne activity problem in progress, determine if the appropriate automatic actions have occurred in accordance with 0AOP-5.0, Radioactive Spills, High Radiation, and Airborne Activity.

Reference:
AOP-05, Radioactive Spills, High Radiation and Airborne Activity

Cog Level:
High

Explanation:
0AOP-05 immediate action for a dropped or damaged fuel assembly is to ENSURE CREVS is in operation.
The dose consequence calculation for the fuel handling accident does not credit the secondary containment or automatic CREVS start, however, it does assume that CREVS is manually initiated within 20 minutes of a dropped/damaged fuel assembly. Based on this analysis, Technical Specifications do not require secondary containment or CREVS automatic initiation instrumentation except during Modes 1, 2, or 3 or during operations with the potential to drain the Reactor vessel. The CREV System is designed to maintain a habitable environment in the CRE for a 30 day continuous occupancy after a DBA without exceeding 5 rem total effective dose equivalent (TEDE).
Knowledge of DBA analysis initial conditions.

Distractor Analysis:

Choice A: Plausible because *PROCESS RX BLDG VENT RAD HIGH* annunciator is easily confused with the auto start for SBGT verifying Auto actions can be confused with Immediate Actions. SBGT start is a supplemental action which will reduce control room dose and 2 Rem TEDE is a site administrative dose limit and can be confused with the actual Dose Analysis from FHA of 2.69 rem TEDE.

Choice B: Plausible because *PROCESS RX BLDG VENT RAD HIGH* annunciator is easily confused with the auto start for SBGT verifying Auto actions can be confused with Immediate Actions. SBGT start is a supplemental action which will reduce control room dose and 5 Rem TEDE is correct.

Choice C: Plausible because CREV is correct and 2 Rem TEDE is a site administrative dose limit and can be confused with the actual Dose Analysis from FHA of 2.69 rem TEDE.

Choice D: Correct Answer, see explanation

SRO Only Basis:
Conditions and limitations in the facility license (43(b)(1)

88. S295026 A2.03 001

An event on Unit One has resulted in the following plant conditions:

Reactor pressure	1000 psig
Reactor Water Level	120 inches
Control Rod Positions	All unknown
APRMs	Downscale
Drywell pressure	3 psig
Supp. Pool pressure	2 psig
Supp. Pool water temp	150° F
Supp. Pool water level	-4 feet

(Reference provided)

Which one of the following identifies the status of the Heat Capacity Temperature Limit (HCTL) and the required procedure for reactor pressure control?

	HCTL	Pressure Control Leg of Procedure
A.	has been exceeded	RVCP
B. ✓	has been exceeded	LPC
C.	has NOT been exceeded	RVCP
D.	has NOT been exceeded	LPC

Feedback

K/A: S295026 A2.03
Ability to determine and/or interpret the following as they apply to SUPPRESSION POOL HIGH WATER TEMPERATURE:
Reactor pressure
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.9/4.0

Objective:
CLS-LP-300-L*05a
Given the PCCP, determine the appropriate actions if any of the following limits are approached or exceeded: Heat Capacity Temperature Limit

Reference:
Heat Capacity Temperature Graph only is given to examinee
PCCP.

Cog Level:
High

Previous Exam - 11/08 Brunswick

Explanation:
HCTL has been exceeded. With rods unknown the operator would be in LPC.

Distractor Analysis:

Choice A: Plausible because rods are unknown, would be in LPC.

Choice B: Correct Answer, see explanation

Choice C: Plausible because HCTL has been exceeded. rods are unknown, would be in LPC

Choice D: Plausible because HCTL has been exceeded.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations (43(b)(5)

89. S295028 A2.05 001

Unit Two is operating at rated power when half of the Drywell (DW) Coolers are lost.

Which one of the following completes the statements below?
(Assume initial DW and Suppression Pool pressures are equal)

Suppression Pool pressure will rise at __(1)__ DW pressure.

If DW Average Air Temperature is not restored to within the LCO limit in __(2)__ hours, the Unit is required to be in Mode 3 within the following 12 hours per TS 3.6.1.4 (Drywell Air Temperature).

A. (1) the same rate as
   (2) 8

B. (1) the same rate as
   (2) 12

C. (1) a slower rate than
   (2) 8

D. (1) a slower rate than
   (2) 12

Feedback

K/A: S295028 A2.05
Ability to determine and/or interpret the following as they apply to HIGH DRYWELL TEMPERATURE :
Torus/suppression chamber pressure: Plant-Specific
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.6/3.8

Objective:
CLS-LP-004-A*15a
Given plant conditions, determine the effects that the following will have on the Primary Containment, Primary Containment Ventilation and Primary Containment Monitoring: Loss of Drywell cooling.

Reference:
SD-04, Primary Containment
Tech Spec 3.6.1.4, Drywell Air Temperature

Cog Level:
High

Explanation:Reduced DW cooling or rising DW temperature results in DW pressure increases whose severity is dependent upon plant conditions. 0AOP-14.0, Abnormal Primary Containment Conditions provides guidance on indications to be monitored and actions to be taken which include verification of cooling system lineups and reductions in power to maintain average temperature below 150°F. Failure to accomplish this may require entry into the 0EOP-02-PCCP Primary Containment Control.
Elevated DW temperature causes DW pressure to rise. As DW pressure rises, SP water level rises causing a rise in SP pressure. Due to the downcomers extending 3 feet below the surface of the SP water level a differential pressure will always exist. Temperature response is different from LOCA response due to steam AND non-condensibles being forced into the SP - steam condensing and non-condensibles collecting in SP air space.

TS 3.6.1.4 (DW Air Temperature) limit of ≤ 150°F, CONDITION A - Drywell average air temperature not within limit, REQUIRED ACTION A.1 Restore drywell average air temperature to within limit has a COMPLETION TIME of 8 hours. If temperature is not restored to ≤ 150°F, CONDITION B, REQUIRED ACTION B.1 Be in MODE 3 has a COMPLETION TIME of 12 hours.

Distractor Analysis:

Choice A: Plausible because SP pressure is changed by the change in SP level only vs pressure, steam, and non-condensibles during a LOCA. The SP air space temperature is in equilibrium with SP water temperature (95°Fmax during normal operations) rising DW pressure would have a direct impact on SP level. However during temperature only (no steam), the DW pressure increase is cushioned by SP water, small changes in SP water level provides small change in SP pressure. 8 hours to restore temperature is correct.

Choice B: Plausible because SP pressure is changed by the change in SP level only vs pressure, steam, and non-condensibles during a LOCA. The SP air space temperature is in equilibrium with SP water temperature (95°Fmax during normal operations) rising DW pressure would have a direct impact on SP level. However during temperature only (no steam), the DW pressure increase is cushioned by SP water, small changes in SP water level provides small change in SP pressure. 12 hours is the time required to get to MODE 3 if not restored within the required Completion Time.

Choice C: Correct Answer, see explanation

Choice D: Plausible because rising at a slower rate is correct and 12 hours is the time required to get to MODE 3 if not restored within the required Completion Time.

SRO Basis:
Application of required actions (Section 3) and surveillance requirements (Section 4) in accordance with rules of application requirements (Section 1). (43(b)(2)

90. S295029 G2.01.07 002

The following plant conditions exist on Unit Two:

- An ATWS with a spurious Group I Isolation has occurred
- HPCI is injecting to the RPV to maintain RPV level
- *SUPPRESSION CHAMBER LVL HI-HI* is in alarm

Which one of the following identifies:

(1) the reason that HPCI is re-aligned from its current suction source and
(2) the procedure that contains the steps to perform the actions to transfer the HPCI suction valves?

A. (1) To prevent pump bearing damage
(2) 2OP-19, High Pressure Coolant Injection System Operating Procedure

B.✓ (1) To prevent pump bearing damage
(2) SEP-10, Circuit Alteration Procedure

C. (1) To prevent HPCI exhaust check valve damage
(2) 2OP-19, High Pressure Coolant Injection System Operating Procedure

D. (1) To prevent HPCI exhaust check valve damage
(2) SEP-10, Circuit Alteration Procedure

Feedback

K/A: 295029 G2.01.07
Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation.
High Suppression Pool Water Level
(CFR: 41.5 / 43.5 / 45.12 / 45.13)

RO/SRO Rating:
4.4/4.7

Objective:
CLS-LP-019-A, Obj. 26g:
Given plant conditions and one of the following events, use plant procedures to determine the actions required to control and/or mitigate the consequences of the event: High Suppression Pool water level

Reference:
2OP-19, HPCI System Operating Procedure
0OI-37.5, Level/Power Control Procedure Basis Document
SUPPRESSION CHAMBER LVL HI-HI APP

Cog Level:
High

Explanation:
Very high lube oil temperatures can result in loss of lubricating qualities in the oil and thus cause damage to the bearings. Suction for HPCI and RCIC is aligned to the Condensate Storage Tank (CST) if it is available. The HPCI automatic suction transfer logic can be defeated to allow this lineup if necessary provided suppression pool temperature is approaching 140°F. Step RC/L-25 of LPC directs maintaining reactor water level with the systems listed in Table 1. "HPCI with suction from CST if available" is listed in Table 1 with the direction to defeat the "Defeat HPCI Hi Suppression Pool level Suction Transfer" if necessary per SEP-10.

Distractor Analysis:

Choice A: Plausible because 2OP-19 contains direction for transferring the HPCI suction from the torus to the CST. (Section 8.9)

Choice B: Correct Answer, see explanation

Choice C: Plausible because RC/L-20 references potential HPCI "equipment damage" which could be interpreted as exhaust check valve damage. 2OP-19 contains direction for transferring the HPCI suction from the torus to the CST. (Section 8.9)

Choice D: Plausible because RC/L-20 references potential HPCI "equipment damage" which could be interpreted as exhaust check valve damage.

SRO Basis:
10 CFR 55.43(b)-5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

91. S295038G 2.02.42 001

Unit Two is operating at rated power when both SJAE Offgas Radiation monitor readings increase by 50% over their previous value.

Which one of the following completes the statement below?

IAW __(1)__ surveillance requirements, the SJAE release rate must be confirmed within the limits within __(2)__.

A. (1) ODCM 7.3.2, Radioactive Gaseous Effluent Monitoring Instrumentation
(2) 4 hours

B. (1) ODCM 7.3.2, Radioactive Gaseous Effluent Monitoring Instrumentation
(2) 12 hours

C✓ (1) Technical Specification 3.7.5, Main Condenser Offgas
(2) 4 hours

D. (1) Technical Specification 3.7.5, Main Condenser Offgas
(2) 12 hours

Feedback

K/A: S295038G 2.02.42
Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
High Off-Site Release Rate
(CFR: 41.7 / 41.10 / 43.2 / 43.3 / 45.3)

RO/SRO Rating:
3.9/4.6

Objective:
CLS-LP-30*08
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR, determine whether given plant conditions meet minimum Technical Specifications, TRM, or ODCM requirements associated with the Condenser Air Removal/Augmented Offgas System.

Reference:
1OI-03.1, Revision 10, Page 44, Item #57 (CODSR)

Cog Level:
High

Explanation:
NOTIFY E&RC to confirm release rate is within limits within 4 hours following a monitor reading increase of greater than or equal to 50% without an accompanying increase in thermal power. SR 3.7.5.1

Distractor Analysis:

Choice A: Plausible because the SJAE Rad Monitor operability is required by ODCM 7.3.2 and 4 hours is correct.

Choice B: Plausible because the SJAE Rad Monitor operability is required by ODCM 7.3.2 and 12 hours is a timeframe for another Required Action in this spec.

Choice C: Correct Answer, see explanation

Choice D: Plausible because TS 3.7.5 is correct and 12 hours is a timeframe for another Required Action in this spec.

SRO Basis:
Application of Surveillance Requirements and timeframe greater than 1 hour.

92. S300000 A2.01 001

The following plant conditions exist on Unit Two due to a malfunction of the Air Dryer:

- *SERVICE AIR PRESS-LOW* is in alarm
- *RB INSTR AIR RECEIVER 2A PRESS LOW* is in alarm
- *RB INSTR AIR RECEIVER 2B PRESS LOW* is in alarm
- Instrument Air pressure is 93 psig and recovering

Based on the above indications, which one of the following identifies:

(1) the status of the Service Air Dryer Bypass Valve, SA-PV-5067, and
(2) the procedure that contains the steps to close the Reactor Building Inboard and Outboard Isolation Valves (BFIVs)?

A. (1) open
(2) 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures

B. (1) open
(2) 2APP-UA-01, *Service Air Press-Low*

C. (1) closed
(2) 0AOP-20.0, Pneumatic (Air/Nitrogen) System Failures

D. (1) closed
(2) 2APP-UA-01, *Service Air Press-Low*

Feedback

K/A: 300000 A2.01
Ability to (a) predict the impacts of the following on the INSTRUMENT AIR SYSTEM and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those abnormal operation:
Air dryer and filter malfunctions
(CFR: 41.5 / 45.6)

RO/SRO Rating:
2.9/2.8

Objective:
CLS-LP-46, 07i:
Given plant conditions, determine if the following automatic actions should occur: Air Dryer is bypassed.

Reference:
RB INSTR AIR RECEIVER 2B PRESS LOW (UA-01 1-2)
SERVICE AIR PRESS LOW (UA-01 5-4)
0AOP-20, Pneumatic (Air/Nitrogen) System Failures

Cog Level:
High

Explanation:
The air dryer malfunction has caused air pressure to lower. The Service Air low pressure alarms comes in at 107 psig. At 105# decreasing the Service Air system isolates, thus the 0 psig indication on Service Air. The alarms for the receivers low pressure come in at 95# and are located in the Reactor Building. With these alarms in the operators are required to close the BFIVs while there is still sufficient air pressure remaining to make the secondary containment isolation valves close in accordance with the AOP supplemental actions.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because the air dryer bypass valve is open, but the guidance for closure of the BFIVs is contained in the AOP or *RB INSTR AIR RECEIVER 2A(B) PRESS LOW* APP.

Choice C: Plausible because the AOP is the correct procedure for closure of the BFIVs, but the air dryer bypass valve would be open (requires system knowledge to know the setpoint for the bypass opening).

Choice D: Plausible because the student may not know the setpoint of the bypass valve opening and the guidance for closure of the BFIVs is contained in the AOP or *RB INSTR AIR RECEIVER 2A(B) PRESS LOW* APP.

SRO Basis:
10 CFR 55.43(b)-5 Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, and emergency situations.

The first part of the question is RO knowledge (setpoint for the auto opening of the air dryer bypass valve the second part is Assessing plant conditions (normal, abnormal, or emergency) and then prescribing a procedure to mitigate, recover, or with which to proceed.

93. S600000G 2.02.37 002

Unit Two is operating at rated power when a fire in the Unit Two Reactor Building 20' North is confirmed.

Which one of the following identifies the impact on safety related equipment and the procedure(s) required to be entered under the above plant conditions?

A fire on the Unit Two Reactor Building 20' North will impact __(1)__ Availability. The Unit CRS is required to enter __(2)__.

A. (1) RHR Loop A
   (2) 0PFP-013, General Fire Plan, only

B.✓ (1) RHR Loop A
   (2) 0PFP-013, General Fire Plan, and 0ASSD-01, Alternative Safe Shutdown Procedure Index

C. (1) RHR Loop B
   (2) 0PFP-013, General Fire Plan, only

D. (1) RHR Loop B
   (2) 0PFP-013, General Fire Plan, and 0ASSD-01, Alternative Safe Shutdown Procedure Index

Feedback

K/A: S600000G 2.02.37
Ability to determine operability and/or availability of safety related equipment.
Plant Fire On Site
(CFR: 41.7 / 43.5 / 45.12)

RO/SRO Rating:
3.6/4.6

Objective:

Reference:
0AP-025, BNP Integrated Scheduling, Section 3.1

Cog Level:
High

Explanation:
2PFP-RB identifies safety related electrical equipment by red or green color coding. (Div I or II respectively) The north side of Unit 2 RB 20' has Div I related MCCs identified which provide power to RHR system motor operated valves. There are no Div II designated MCCs identifies for the same location.
Attachment 2 of PFP-013 "Senior Control Operator Fire Actions" directs the SCO to enter PFP-013 and ASSD-01.

Distractor Analysis:

Choice A: Plausible because RHR Loop A is correct and 0PFP-013 directs entry into both PFP-013 and ASSD-01.

Choice B: Correct Answer, see explanation

Choice C: Plausible because RHR Loop B can be impacted by a fire in RB South and 0PFP-013 directs entry into both PFP-013 and ASSD-01.

Choice D: Plausible because RHR Loop B can be impacted by a fire in RB South and part 2 is correct.

SRO Basis:
Knowledge of administrative procedures that specify implementation, and/or coordination of plant normal procedures.

94. SG2.01.05 001

Which one of the following choices completes the statement below?

If the ASSD staffing composition is less than the minimum required, establish an __(1)__ IAW __(2)__.

A. (1) ASSD Impairment
(2) 0PLP-1.5, Alternative Shutdown Capability Controls

B. (1) LCO for T.S. 5.2.2, Facility Staff Organization,
(2) 0PLP-1.5, Alternative Shutdown Capability Controls

C. (1) ASSD Impairment
(2) 0OI-01.01, BNP Conduct of Operations Supplement

D. (1) LCO for T.S. 5.2.2, Facility Staff Organization,
(2) 0OI-01.01, BNP Conduct of Operations Supplement

Feedback

K/A: SG2.01.05
Conduct of Operations
Ability to use procedures related to shift staffing, such as minimum crew complement, overtime limitations, etc.
(CFR: 41.10 / 43.5 / 45.12)

RO/SRO Rating:
2.9/3.9

Objective:
CLS-LP-304-M*13m
Given ASSD procedures and plant conditions that require use of ASSD procedures, determine the following: The manpower required to support the ASSD actions.

Reference:
ASSD-00, ASSD Users Guide, Section 5.3.3

Cog Level:
Low

Explanation:
The ASSD staffing composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore requirements to within the minimum requirements of the shift ASSD staffing. If the ASSD staffing composition is less than the minimum required, establish an Alternative Safe Shutdown Impairment in accordance with 0PLP-1.5, Alternative Shutdown Capability Controls, and 0FPP-020, Impairment Notification.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because an impairment is the same as LCO (0OI-01.01) but impairments are not established against TS 5.2.2 and ASSD User Guide is correct.

Choice C: Plausible because ASSD impairment is correct and 0OI-01.01 provides staffing requirements for TS 5.2.2 but directs use of 0ASSD-00 procedure use for required staffing.

Choice D: Plausible because Plausible because an impairment is the same as LCO (0OI-01.01) but impairments are not established against TS 5.2.2 and 0OI-01.01 provides staffing requirements for TS 5.2.2 but directs use of 0ASSD-00 procedure use for required staffing.

SRO Basis:
Requires knowledge of TS 5.2.2 Facility Staff Organization - and prescribes the procedure required for guidance during periods of ASSD minimum complement not maintained.

95. SG2.01.42 001

Which one of the following core loading sequences establishes a neutronic bridge as described in 0FH-11?

Four fuel bundles are loaded around __(1)__, then fuel is loaded in all fuel cells in a line between SRMs __(2)__.

A. (1) SRMs A and D only
   (2) A and D

B. (1) SRMs B and D only
   (2) B and D

C. (1) each of the four SRMs
   (2) A and D

D. (1) each of the four SRMs
   (2) B and D

Feedback

K/A: SG2.01.42
Conduct of Operations
Knowledge of new and spent fuel movement procedures.
(CFR: 41.10 / 43.7 / 45.13)

RO/SRO Rating:
2.5/3.4

Objective:

Reference:
0FH-11, Refueling, Section 4.37

Cog Level:
High

Explanation:
Provide ENP-24-12, Figure 1 as a reference

From FH-11, 4.37

To help ensure that an unmonitored criticality will not occur, control rod withdrawal is not allowed during the core reload sequence until a neutronic bridge is established. The neutronic bridge ensures that two SRMs are neutronically coupled, thus monitoring the loaded area of the core. The reload sequence has three basic steps. Four fuel bundles are loaded around each of the four SRMs, the neutronic bridge is established and a spiral reload of the other fuel bundles completes the sequence. The neutronic bridge is established by loading fuel in all fuel cells in a line between two SRMs. These SRMs must be on opposite sides of the core and the line of loaded fuel cells must intersect the center of the core (A to D would not intersect the center, B to D would).

Distractor Analysis:

Choice A: Plausible because loading fuel around 2 SRMs and a line between them would establish a neutron bridge (between those 2 SRMs) but not IAW 0FH-11 and A&D are adjacent.

Choice B: Plausible because loading fuel around 2 SRMs and a line between them would establish a neutron bridge (between those 2 SRMs) but not IAW 0FH-11 and A&D are adjacent.

Choice C: Plausible because loading fuel around all SRMs is correct but A&D are adjacent.

Choice D: Correct Answer, see explanation

SRO Basis:
10CFR55.43.6 Procedures and limitations involved in initial core loading, alterations in core configuration, control rod programming, and determination of various internal and external effects on core reactivity.
10CFR55.43.7 Fuel handling facilities and procedures.

96. SG2.02.15 001

With Unit Two operating at power, Annunciator *RCIC TURBINE STM LINE DRN POT LEVEL HI* alarms and the RO observes the E51-F054, F025, & F026 indications on Panel P601 are green and the valves cannot be re-opened.

Which one of the following completes the statements below?

(Reference provided)

These valves are closed due to loss of __(1)__ .
After 5 minutes the appropriate actions in the annunciator procedure have been taken and the system __(2)__.

A. (1) pneumatics
   (2) remains Operable

B.✓ (1) pneumatics
   (2) is Inoperable

C. (1) DC Power
   (2) remains Operable

D. (1) DC Power
   (2) is Inoperable

Feedback

K/A: SG2.02.15
EQUIPMENT CONTROL
Ability to determine the expected plant configuration using design and configuration control documentation, such as drawings, line-ups, tag-outs, etc.
(CFR: 41.10 / 43.3 / 45.13)

RO/SRO Rating:
3.9/4.3

Objective:
CLS-LP-016*15e
Given plant conditions, predict the RCIC System response to the following conditions: DC power failure.

Reference:
2APP A-03 3-5, Page 44

Cog Level:
High

Explanation:
Valves fail closed on loss of DC power or Pneumatics, however with a loss of power, position indication on P601 will also be lost. Per APP A-03, 3-5 - If either E51-F025 or E51-F026 has been failed closed for more than 5 minutes, perform the following:

a. Close Turbine Trip and Throttle Valve, E51-V8, to prevent water hammer damage from a RCIC auto start.
b. If RCIC **must** be started, proceed to OP-16.
this would still make RCIC available for use per the procedure but it is inoperable because it will not auto start as required.

This will make the RCIC system inoperable but available to be restarted per the procedure.

Distractor Analysis:

Choice A: Plausible because loss of pneumatics only is correct and the system will not start in auto when required, but could be manually started.

Choice B: Correct Answer, see explanation

Choice C: Plausible because a loss of power will cause valves to fail closed, but with loss of power position indication will be lost and the system will not start in auto when required.

Choice D: Plausible because pneumatics and power will cause valves to fail closed, but with loss of power position indication will be lost and it is available to start per the procedure which makes it available.

SRO Basis:
Assessment of facility conditions and selection of appropriate procedures during normal, abnormal, emergency conditions.

97. SG2.02.22 002

Unit One is operating at 88% power with the following conditions:

Jet Pump Flow Loop A (B21-R611A)	29 Mlbs/hr
Jet Pump Flow Loop B (B21-R611B)	33 Mlbs/hr
Total Core Flow (U1CPWTCF)	62 Mlbs/hr

Which one of the following completes the statements below IAW T.S. 3.4.1, Recirculation Loops Operating, and Bases?

The current Jet Pump Flow Mismatch ___(1)___.
When Jet Pump Flows are not matched within limits, then the loop with the ___(2)___ must be considered not in operation.

A. (1) is within limits
   (2) lower flow

B. (1) is within limits
   (2) higher flow

C. (1) is not within limits
   (2) lower flow

D. (1) is not within limits
   (2) higher flow

Feedback

K/A: SG2.02.22
Equipment Control
Knowledge of limiting conditions for operations and safety limits.
(CFR: 41.5 / 43.2 / 45.2)

RO/SRO Rating:
4.0/4.7

Objective:
CLS-LP-002*34
Given plant conditions and Technical Specifications, including the Bases, TRM, ODCM, and COLR determine the required action(s) to be taken in accordance with Technical Specifications associated with the Reactor Recirculation System. (SRO/STA only)

Reference:
Unit 1 Tech Spec 3.4.1, Recirculation Loops Operating and BASES

Cog Level:
High

Explanation:
Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied.
Jet pump loop flow mismatch should be maintained within the following limits:
- jet pump loop flows within 10% (maximum indicated difference 7.5 $x10^6$ lbs/hr) with total core flow less than 58 $x10^6$ lbs/hr
- jet pump loop flows within 5% (maximum indicated difference 3.5 $x10^6$ lbs/hr) with total core flow greater than or equal to 58 $x10^6$ lbs/hr

Distractor Analysis:

Choice A: Plausible because flow mismatch is within limits for lower reactor power level.

Choice B: Plausible because flow mismatch is within limits for lower reactor power level and because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response

Choice C: Correct Answer, see explanation

Choice D: Plausible because the belief that the higher flow loop will experience excessive vibration could cause them to select the "higher flow" response

SRO Basis:
Application of Required Actions and Knowledge of TS Bases.

98. SG2.03.11 001

Following a small steam line break in the drywell plant conditions are as follows:

Drywell pressure:	20 psig and rising
Drywell Temperature:	305°F
Drywell hydrogen:	1.8%
Suppression Chamber hydrogen:	1.5%
Suppression Chamber level:	+5 inches

Which one of the following identifies the procedure required IAW PCCP?

A. SEP-01 Section 1, Venting Primary Containment Irrespective of Off Site Release rate

B. ✓ SEP-01, Section 2, Venting Primary Containment via the Suppression Chamber within Site Release Rate Limit

C. SEP-01, Section 3, Venting Primary Containment via the Drywell within Site Release Rate Limit

D. 0EDMG-003, Containment Venting Under Conditions of Extreme Damage Irrespective of Off Site Release Rates

Feedback

K/A: SG2.03.11
Radiation Control
Ability to control radiation releases.
(CFR: 41.11 / 43.4 / 45.10)

RO/SRO Rating:
3.8/4.3

Objective:
CLS-LP-300-L*08d
Given the Primary Containment Control Procedure and plant conditions, determine if the following actions are required: Venting the primary containment IRRESPECTIVE of radioactivity release rate limits

Reference:
0OI-37.8, Primary Containment Control Procedure Basis Document

Cog Level:
High

Explanation:
Following the H2 leg of the PCCP with the given conditions will drive you to step PC/H-14 which directs you to "Vent the Suppression Chamber per Section 2 of Primary Containment Venting (EOP-01-SEP-01)

Distractor Analysis:

Choice A: Plausible if examinee determines that PCPL-A is being challenged out of the pressure leg of the PCCP

Choice B: Correct Answer, see explanation

Choice C: Plausible if examinee determines that the appropriate vent path is from the drywell vs. the suppression chamber.

Choice D: Plausible because EDMG-03 provides alternative venting strategies which may be directed from SEP-01 in needed.

SRO Basis:
Detailed knowledge of diagnostic steps and decision points in the EOPs that involve transitions to emergency contingency procedures.

99. SG2.04.17 002

Which one of the following identifies the bases for the Minimum Number of SRVs Required for Emergency Depressurization and the required procedure utilized if this number of SRVs open cannot be achieved?

The Minimum Number of SRVs Required for Emergency Depressurization is based on the low pressure ECCS system with the lowest head being capable of making up the SRV steam flow at the __(1)__.

If the minimum number of SRVs cannot be opened while performing LPC, then __(2)__ is required to be entered.

A. (1) Reactor Flooding Pressure
   (2) SAMG-01, SAMG Primary Containment Flooding Procedure

B. (1) Reactor Flooding Pressure
   (2) 0EOP-01-AEDP, Alternate Emergency Depressurization Procedure

C. (1) Minimum Alternate Reactor Flooding Pressure
   (2) SAMG-01, SAMG Primary Containment Flooding Procedure

D. (1) Minimum Alternate Reactor Flooding Pressure
   (2) 0EOP-01-AEDP, Alternate Emergency Depressurization Procedure

Feedback

K/A: SG2.04.17
Emergency Procedures / Plan
Knowledge of EOP terms and definitions.
(CFR: 41.10 / 45.13)

RO/SRO Rating:
3.9/4.3

Objective:
CLS-LP-300-H*002
Given plant conditions and the Emergency Operating Procedures, determine if execution of the Alternate Emergency Depressurization Procedure is required.

Reference:
0EOP-01-UG, Attachment 5 (Definitions)
RVCP

Cog Level:
High

Explanation:
The Minimum Number of SRVs Required for Emergency Depressurization (5) is defined to be the least number of SRVs which correspond to a Minimum Alternate Reactor Flooding Pressure sufficiently low that the ECCS with the lowest head will be capable of making up the SRV steam flow at the corresponding Minimum Alternate Reactor Flooding Pressure. If the number of SRVs specified cannot be opened, the reactor must be depressurized by other means. A list of alternate systems that can be used for depressurizing the reactor is included in the Alternate Emergency Depressurization Procedure, EOP-01-AEDP.

Distractor Analysis:

Choice A: Plausible because Minimum Reactor Flooding Pressure is easily confused with Minimum Alternate Reactor Flooding Pressure and Primary Containment Flooding requires exiting all EOPs which is wrong for the given conditions.

Choice B: Plausible because Minimum Reactor Flooding Pressure is easily confused with Minimum Alternate Reactor Flooding Pressure and AEDP is correct.

Choice C: Plausible because Minimum Alternate Reactor Flooding Pressure is correct and Primary Containment Flooding requires exiting all EOPs which is wrong for the given conditions.

Choice D: Correct Answer, see explanation

SRO Basis:
Detailed knowledge of diagnostic steps and decision points in the EOPs that involve transitions to emergency contingency procedures.

100. SG2.04.35 002

An ATWS has occurred on Unit Two:

ARI has been actuated
No blue lights are lit on the Full Core Display
Suppression Pool Temperature is 112° F
The 2A SLC pump has a red light indication
The 2B SLC pump has a green light indication
The SLC A Squib Valve Continuity white light is lit
The SLC B Squib Valve Continuity white light is extinguished

Which one of the following identifies the procedure that an AO is required to perform?

A. ✓ Perform LEP-02, Alternate Control Rod Insertion, Section 2.

B. Perform LEP-02, Alternate Control Rod Insertion, Section 6.

C. Perform LEP-03, Alternate Boron Injection, Section 2.

D. Perform LEP-03, Alternate Boron Injection, Section 3.

Feedback

K/A: SG2.04.35
Emergency Procedures / Plan
Knowledge of local auxiliary operator tasks during an emergency and the resultant operational effects.
(CFR: 41.10 / 43.5 / 45.13)

RO/SRO Rating:
3.8/4.0

Objective:
CLS-LP-300-J*005
Given plant conditions and the Local Emergency Procedures, determine which sections of the Alternate Control Rod Insertion Procedure should be utilized for Control Rod Insertion (EOP-01-LEP-02).

Reference:
0EOP-01-LEP-02, Alternate Control Rod Insertion

Cog Level:
High

Explanation:
Based on the conditions given, determines that scram valves have not opened (no blue lights on full core display) and that Boron is injecting with A pump running (red light on) and B squib valve opened (white light extinguished) so LEP-03 is not required. The pumps discharge into a common header before going to the squib valves. Requires assessment of alternate control rod insertion sections and determines venting the scram air header is appropriate.

Distractor Analysis:

Choice A: Correct Answer, see explanation

Choice B: Plausible because venting of the over piston area will insert the control rods but would be the inappropriate decision for rod insertion given the conditions. The operational effect is reactor shutdown with control rod insertion.

Choice C: Plausible because suppression pool temperature is greater than 110° F and boron injection is required. With A pump running but the A squib valve not open and no B pump a common misconception is that SLC flow will not occur to the Reactor. this would be correct under different conditions in the stem. The operational effect is reactor shutdown with boron injection.

Choice D: Plausible because suppression pool temperature is greater than 110° F and boron injection is required. With A pump running but the A squib valve not open and no B pump a common misconception is that SLC flow will not occur to the Reactor. this would be correct under different conditions in the stem. The operational effect is reactor shutdown with boron injection.

SRO Basis:
Assessing plant conditions and prescribing a section of a procedure with which to proceed.