*QNUM 001
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000007K301
QUESTION
The following Unit 1 plant conditions exist:

- Unit 1 has experienced a reactor trip and SI.
- Containment pressure is 27 psig.
- RCS pressure is 300 psig.
- Seven of Eight SX Cooling Tower Fans are running in High Speed. 0A fan will NOT start in High Speed.

Which ONE of the following actions is required per 1BEP-0, Reactor Trip or Safety Injection, when aligning the SX Cooling Towers?

a. OPEN all EIGHT riser valves.

b. Restart 0A fan in Low Speed

c. CLOSE all FOUR Hot Water Basin Bypass valves.

d. Ensure that ONLY the bypass valve associated with the non-running fan is CLOSED

Answer
c.
Reference
1BEP-0, Reactor Trip or Safety Injection
I1-EP-XL-01, 1BEP-0, Reactor Trip or Safety Injection
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the reasons for the following as they apply to a reactor trip: Actions contained in EOP for reactor trip.
Explanation

a. Is incorrect, because the riser valve associated with the failed fan will need to be closed.
b. Is correct, low speed is not acceptable.
c. IS correct per 1BEP-0, Reactor Trip or Safety Injection.
d. Is incorrect. All four bypass valves should be closed.

*QNUM	002
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	000008A101

QUESTION

Unit 1 was at 100% power with all systems normally aligned when annunciator 1-12-B2, PZR PORV OR SAF VLV OPEN, alarms. The following indications are current:

- Actual PZR pressure is 2100 psig and lowering
- Channel 1PT-455 indicates 2500 psig
- PZR level is 62% and rising
- PRT temperature, pressure and level are rising
- All PZR Safety Valve indicator lights are GREEN

Action(s) to mitigate this transient is/are to …

a. close the PZR PORV block valve(s) for affected PORV(s).

b. manually trip the reactor and actuate SI.

c. verify insertion of control rods at 48 steps per minute.

d. manually trip the reactor, but DO NOT manually actuate SI.

ANSWER

a.

REFERENCE

Horse Notes RY-2, PZR Pressure Control, Rev. 2;
RY-1, Pressurizer, Rev. 3;
Lesson Plan, Pressurizer (RY), Rev. 6, Attachment B;
BAR 1-12-B2, PZR PORV OR SAF VLV OPEN, Rev. 4;
1BOA INST-2, Rev. 103

New

Higher

K/A Ability to operate and / or monitor the PZR spray block valve and PORV block valve as applied to the PZR Vapor Space Accident.

EXPLANATION

a. correct
b. incorrect, actions for inability to maintain PZR pressure (1BOA INST-2, Rev. 103).
c. incorrect, because there is no ATWS in process and there is no trip required.
d. incorrect. Partial action for inability to maintain PZR pressure (1BOA INST-2, Rev. 103)

*QNUM 003
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000009A213
QUESTION
Given the following plant conditions for Unit 1:

- Indicated charging flow is 152 gpm.
- Letdown flow is 120 gpm.
- Total seal injection flow is 32 gpm.
- Total seal return flow is 12 gpm.
- Pressurizer level is stable.

Does a primary leak exist, and if so, what is the size of the leak?

There is…

a. NO primary leak.

b. a 20 gpm leak.

c. a 32 gpm leak.

d. a 52 gpm leak.

ANSWER
b.
REFERENCE
Horse Notes CV-1, CVCS, Rev. 2; RCP-1, RCP Seal Package, Rev. 2
New
Higher
EXPLANATION
There is total charging flow of 152 gpm Letdown flow is 120 gpm and seal return flow is 12 gpm. The difference between water entering the RCS and leaving the RCS via designed systems is 20 gpm. With Pzr level Pzr level stable, there is a 20 gpm leak.

Choice A is arrived at by subtracting seal injection and letdown from charging flow.

Choice C is arrived at by subtracting 120 gpm letdown from 152 gpm charging.

Choice D is arrived at by adding seal injection to charging and subtracting letdown and seal return.

*QNUM 004
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000022 2.1.30
QUESTION
Unit 1 is operating at 100% power with all systems normally aligned:

- 1A CV pump running
- PZR level at 60%
- Charging flow 132 gpm
- Seal injection flows 11 gpm per RCP
- 1CV121, Centrifugal Charging Pump Flow Control Valve, in MANUAL
- 1CV182, Charging Header Back Pressure Control Valve, at 52% demand
- 1A CV impeller begins to slowly degrade

In order to maintain pressurizer level at 60% and RCP seal injection flows at 11 gpm, operators must…

a. throttle close 1CV121 AND 1CV182.

b. throttle open 1CV182 ONLY.

c. throttle close 1CV121 AND throttle open 1CV182.

d. throttle open 1CV121 AND maintain 1CV182 at 52% demand.

ANSWER
d.
REFERENCE
Horse Notes CV-1, CVCS, Rev. 2
LP: Chemical and Volume Control System, Rev. 8, Section III
System Description, A.2.e and .f
Higher
New
K/A Ability to locate and operate components, including local controls
EXPLANATION
a. Is incorrect because 1CV121 needs to be opened to maintain PZR level with a degrading pump.
b. Is incorrect because if 1CV121 is not opened also, opening 1CV182 will not additional flow.
c. Is incorrect because 1CV121 needs to be opened to maintain PZR level with a degrading pump.
d. Is correct because opening 1CV121 will maintain PZR level with a degrading pump. With 1CV182 at a constant demand, restoring charging flow will also restore seal injection flow to 11 gpm.

*QNUM 005
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000015/17K102
QUESTION
Unit 1 is operating at 100% power with all systems normally aligned. During the shift, the following 1A RCP indications have been observed:

- Motor amps slowly rising
- Shaft vibrations slowly rising
- Seal inlet temperature 116 deg F, stable
- #1 seal outlet temperature 132 deg F, stable
- High range seal leakage recorder reads 4.2 gpm, stable
- #1 seal delta P greater than 400 psid (pegged high)
- 1A RCP loop flow slowly lowering

These cumulative conditions indicate the…

a. #1 seal is failing.

b. #2 seal is failing.

c. shaft has sheared.

d. thrust bearing is failing.

ANSWER
d.
REFERENCE
Horse Notes RCP-1, RCP Seal Package, Rev. 2
LP: Reactor Coolant and Reactor Coolant Pumps, Rev. 2, Section III, System Description, A.2.a.9)c); B.2.,3.,5.
Section IV, System Operation, B.1.d, B.2.e, B.2.j
BOA RCP-1, Reactor Coolant Pump Seal Failure, Rev. 102
BOP RC-1
New
Higher
K/A Knowledge of the operational implications of concepts for Consequences of an RCPS failure as applied to RCP Malfunctions (Loss of RC Flow).
EXPLANATION
a. Seal indication is normal.
a. Seal indication is normal.
c. Sheared shaft would result in immediate drop in flow and rise in pump vibration.
d. As the thrust bearing begins to fail, bearing friction begins to rise resulting in rising motor amps to overcome the increased friction. Increased friction would also lead to lowering flow and rising pump vibration. Only "D" would account for all listed parameters.

*QNUM 006
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000029K105
QUESTION
Based upon time in core life, which of the following statements describes the transient expected during an ATWS?

a. At BOL conditions, the MORE negative doppler temperature coefficient results in higher peak power.

b. At BOL conditions, the LESS negative moderator temperature coefficient allows higher peak temperatures and pressures.

c. At EOL conditions, the LESS negative doppler temperature coefficient allows higher peak temperatures and pressures.

d. At EOL conditions, the LESS negative moderator temperature coefficient results in higher peak power.

ANSWER
b.
REFERENCE
Horse Notes TH-1, Rx Theory-1, Rev. 0; TH-2, Rx Theory-1, Rev. 0.
New
Higher
K/A Knowledge of the operational implications of the concept for definition of negative temperature coefficient as applied to large PWR coolant systems and ATWS.
EXPLANATION
a. FTC (doppler temperature coefficient) becomes MORE negative over core life (ie, LESS negative at BOL) due to buildup of Pu-240, which has more and higher resonance peaks than U-238.
b. Moderator temperature coefficient (MTC) becomes MORE negative over core life (ie, LESS negative at BOL) due to increased core buckling (ie, more neutron leakage), fuel depletion,
c. FTC is the change in reactivity per degree change in effective fuel temperature and becomes MORE negative over core life due to buildup of Pu-240 which has more and higher resonance peaks than U-238.
d. Moderator temperature coefficient (MTC) becomes MORE negative over core life due to increased core buckling (ie, more neutron leakage), fuel depletion,

*QNUM 007
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000038K304
QUESTION
The CAUTION before Step 1 of BEP ES-3.1, Post SGTR Cooldown Using Backfill, warns that the RCP in the ruptured loop should NOT be started first. Which of the following explains the reason for this CAUTION?

Starting the ruptured loop's RCP first could cause …

a. an internal pressure surge resulting in further failure of damaged tubes.

b. a slug of cold water to pass through the core and cause a return to criticality.

c. a slug of unborated water to pass through the core and cause a return to criticality.

d. a rapid reactor vessel cooldown that would challenge the Integrity critical safety function.

ANSWER
c.
REFERENCE
1BEP ES-3.1, Post-SGTR Cooldown Using Backfill Unit 1, Rev. 101; ILT Simulator Lesson Plan BEP-3 Series, SGTR, Rev. 5, Section V, Post-SGTR Cooldown Using Backfill, CAUTION, p. 59 of 123.
New
Higher
K/A Knowledge of the reasons for the Actions Contained in EOP for RCS Water Inventory Balance, SGTR, and Plant Shutdown Procedures as applied to the SGTR.
EXPLANATION
a. Is incorrect, because the backfill water would be at or less than RCS temperature.
b. Cold water is a PTS concern.
c. Initiation of forced RCS flow by starting an RCP in a loop with an intact steam generator will ensure mixing of the diluted water in the stagnant loop with the borated water in the other RCS loops such that localized dilution in the core region is precluded. Refer to ERG Maintenance Item DW-89-041 for a detailed discussion of the possibility of a localized boron dilution transient
d. Is incorrect, because the intact SGs are used for cooldown. The ruptured SG is not a significant heat sink.

*QNUM 008
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000040A107
QUESTION
For a steam line break inside containment, all of the following will indicate a RISE in value on the Narrow Range SPDS iconic, EXCEPT…

a. Subcooling Margin.

b. Net Charging.

c. Containment Air Temperature.

d. RVLIS.

ANSWER
d.
REFERENCE
???
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Steam Pressure and Flow Rates Via Computer, Safety Parameter Display System, and other Indications
EXPLANATION

a. is incorrect, Subcooling will rise as RCS temperature lowers faster than pressure lowers.
b. is incorrect, Charging will rise as Pzr level lowers.
c. is incorrect, high energy steam line break inside containment
d. is correct-RVLIS is at 100% so cannot rise; also RVLIS is on Wide Range iconic, not NR.

*QNUM 009
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000054A207
QUESTION
Unit 1 was operating at 100% power preparing to enter coast down for an end of cycle refueling outage. The 1A FW pump was OOS with all other systems normally aligned.

An inadvertent FW isolation occurred. The following conditions are present 60 seconds after the FW isolation:

- PR NIs are 0%
- IR SUR is -0.5 dpm
- MSIVs are OPEN
- Annunciator 1-18-C3 EMER TRIP HDR PRESS LOW TRIP is NOT lit
- SG 1A -1D Train B AF flow is greater than 160 gpm per SG
- SG WR levels are 25%, lowering
- RCS temperature is 480 deg F, lowering

These cumulative conditions indicate …

a. AF is NOT flowing to the SGs. RCS pressure will exceed the accident analysis limits during the event.

b. BOTH reactor and turbine are tripped. RCS pressure will NOT exceed the accident analysis limits during the event.

c. BOTH reactor and turbine are NOT tripped. RCS pressure will exceed the accident analysis limits during the event.

d. the reactor is tripped, but the turbine is NOT tripped. RCS pressure will NOT exceed the accident analysis limits during the event.

ANSWER
d.
REFERENCE
BAR 1-18-C3, EMER TRIP HDR PRESS LOW TRIP, Rev. 1; 1BFR-S.1, Response to Nuclear Power Generation/ATWS Unit 1, Rev. 102; LP Auxiliary Feedwater System, Rev 5; ILT BFR S Series Subcriticality, Rev.3.
New
Higher
K/A Ability to determine and interpret the Occurrence of Reactor and/or Turbine Trip as applied to the Loss of MFW.
EXPLANATION
a. Is incorrect because the FW isolation does not impact AF flow. Also, indicated AF flow is greater than 160 gpm per SG.
b. Is incorrect because the turbine has not tripped on the reactor trip.
c. Is incorrect because the turbine has not tripped on the reactor trip. Also, RCS pressure is lowering due to the cooldown created by the turbine not being tripped.
d. Is correct. Also, RCS pressure is lowering due to the cooldown created by the turbine not being tripped.

*QNUM 010
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000055 2.4.50
QUESTION
The reactor has tripped from 100% power due to a Loss of All AC Power, the EDGs failed to reenergize the 4160 volt ESF buses and the following plant indications are noted.

- RCS pressure is 1575 psig and lowering.
- RCS hot leg temperatures indicate 604 deg F and rising.
- Pressurizer level indicates 90% and rising.
- A steam bubble exists in the reactor vessel.

Which ONE of the following is most likely to cause these indications and what actions should be taken to mitigate the situation?

a. A pressurizer PORV has failed in the open position. Manually or locally close the PORV block valve when power is available.

b. A Steam Generator PORV has failed open. An operator needs to be dispatched to locally isolate the failed open PORV.

c. Lack of an adequate heat sink has caused a bubble to form in the reactor vessel. Reduce Steam Generator pressures to lower RCS temperatures to reduce vessel bubble.

d. The steam dump controller has failed with all steam dumps fully closed. Send operators to locally throttle the steam dumps to control the plant heatup rate.

*ANSWER
a.
*REFERENCE
1BCA-0.0, 0.1, 0.2, Loss of All AC Power, Unit 1, Rev. 105, page 4 of 97
New
Higher
EXPLANATION
a. Is correct because there are indications of a PZR vapor space leak.
b. Is incorrect because an open S/G PORV would lower RCSpressure and hot leg temperatures.
c. Is correct because there are indications of a PZR vapor space leak.
d. Is correct because there are indications of a PZR vapor space leak. Also, there is no method to locally throttle steam dump valves

*QNUM 011
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000056K101
QUESTION
Category Statement:
After a Loss of All AC Power, the Operations crew is in procedure 1BCA-0.1, Loss of All AC Power Recovery Without SI Required. Verification of natural circulation cooling can be confirmed by which ONE of the following?

a. RCS subcooling is adequate, SG pressure dropping, RCS hot leg temperature rising.

b. SG pressure dropping, RCS hot leg temperature stable, RCS cold leg temperature rising.

c. RCS subcooling is adequate, RCS hot leg temperature dropping, Average of ten highest core exit thermocouples dropping.

d. SG pressure rising, RCS hot leg temperature dropping, RCS cold leg temperature rising.

ANSWER
c.
REFERENCE
1BCA-0.1, Loss of all AC Power Recovery without SI Required
New
Fundamental
EXPLANATION

a. Is incorrect because hot leg temperature should be stable or dropping.
b. Is incorrect because RCS cold leg temperature should at saturation temperature for SG pressure.
c. Is correct.
d. Is incorrect because SG pressure should be stable or dropping.

*QNUM 012
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000057K301
QUESTION
Why is LOCAL operator action required for a loss of Instrument Bus 214 that resulted in a reactor trip and safety injection?

a. The flow from 2B Auxiliary Feedwater train to all Steam Generators will be excessive.

b. The flow from 2B Auxiliary Feedwater train to all Steam Generators will be too low.

c. 2A Auxiliary Feedwater Pump will need to be LOCALLY started.

d. 2B Auxiliary Feedwater Pump will need to be LOCALLY started.

ANSWER
b.
REFERENCE
???
New
Fundamental
K/A Knowledge of the reasons for the following responses as they apply to the Loss of Vital AC Instrument Bus: Actions contained in EOP for loss of vital ac electrical instrument bus.
EXPLANATION
A is incorrect, the MCR controllers will fail to 0 demand, reducing B train flow to 0.
B is correct, the MCR controllers will fail to 0 demand, reducing B train flow to 0.
C is incorrect, the 2A AF pump will auto start on SI
D is incorrect, the 2B AF pump can be manually started from the MCR.

COMMENTS

*QNUM 013
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000058A101
QUESTION
Previously, 125 VDC Bus 211 was crosstied to Bus 111 due to equipment problems with Bus 111 Battery and Charger. Bus 111 Battery and Charger are Out of Service.

Presently,

- U-1 is in MODE 3.
- U-2 is in MODE 1.

Bus 111 conditions are:

- Crosstie loading due to the loading on Bus 111 is 183 Amps.
- Voltage on Bus 111 is 121 VDC.
- Then, a ground of 50 volts is detected on Bus 111.

Based upon the above conditions, which ONE of the following actions would be CORRECT?

a. Parameters on Bus 111 are normal and within limits. No action is necessary.

b. Enter into BOP, DC-15, DC Ground Isolation, due to an unexpected ground detected on Bus 111.

c. Shed non-essential loads from Bus 111 to lower Amperage to below 180 Amps to meet cross-tie loading restrictions.

d. Disconnect Bus 111 from Bus 211 in accordance with BOP DC-7, 125 VDC ESF Bus Crosstie/Restoration to ensure that the ground does not adversely affect loads on the operating unit.

ANSWER
a.
REFERENCE
BOP DC-7, 125 VDC ESF Crosstie/Restoration
New
Higher
K/A Ability to operate and/or monitor the following as they apply to the Loss of DC Power: Cross-tie of the affected dc bus with the alternate supply.
Proposed references to be provided to applicants during examination: None
EXPLANATION

a. Is correct.
b. Is incorrect, because a 50 Volt ground is expected due to the other unit's ground detector.
c. Is incorrect, because cross-tie loading restrictions are at 200 Amps.
d. Is incorrect, because a 50 Volt ground is expected due to the other unit's ground detector.

COMMENTS

*QNUM	014
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	000062A201

QUESTION
Given the following conditions:

- A unit 1 SX leak in the Aux building has been identified and isolated from the main control room.
- A clearance order is being prepared by an RO.
- A Safety Injection has just occurred on unit 1.

The leak will remain isolated if it is located in which of the following components?

a. Unit 1 Reactor Containment Fan Coolers

b. 1B Containment Chiller Condenser

c. 1A Emergency Diesel Generator

d. 1B Auxiliary Feedwater Pump

ANSWER
b.
REFERENCE
I1-SX-XL-01, Essential Service Water System (Training Plan)
New
Fundamental
Proposed references to be provided to applicants during examination: None
EXPLANATION
a. is incorrect, 1SX016 and 027 will open
b. is correct, 1SX114 and 116 will remain closed.
c. is incorrect, 1SX169 will open
d. is incorrect, the 1B AF pump SX flow cannot be isolated from the control room.
1SX173 and 1SX178 are manual valves.

*QNUM 015
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000065 2.4.2
QUESTION
The operators are performing 1BOA SEC-4, Loss of Instrument Air, with all other conditions normal and the unit at 90% load.

If the reactor is NOT manually tripped as instrument air pressure continues to lower, then the initial automatic reactor trip is the result of...

a. Main Feedwater Regulating Valves FW510-540 failing CLOSED.

b. Letdown Orifice Isolation Valves CV8149A, B, and C failing CLOSED.

c. Main Feedwater Regulating Valves FW510-540 failing OPEN.

d. Centrifugal Charging Pumps flow control valve, CV121, failing OPEN.

ANSWER
a.
REFERENCE
BOA SEC-4, Loss of Instrument Air
BEP-0, Reactor Trip or Safety Injection
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of how the Event-Based Emergency/Abnormal Operating Procedures are used in Conjunction with the Symptom-Based EOPs.
EXPLANATION
a. Is correct because the FW valves fail closed leading quickly to a reactor trip on Lo-2 SG level within a few minutes.
b. Is incorrect because the loss of letdown flow leads to a high PZR level reactor trip after at least 30 minutes.
c. Is incorrect because the FW valves fail closed.
d. Is incorrect because the 1CV121 valve failing open causes a reactor trip on high PZR level after at least 30 minutes.

*QNUM 016
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A W/E04K201
QUESTION
Given the following conditions:

- An inadvertent Safety Injection signal tripped the reactor.
- All equipment responded as required with the exception that 1CV8160, Letdown Line Inboard Containment Isolation Valve, failed to close.
- Pressurizer level fell to 25% and is now at 30% and rising.
- Pressurizer Pressure is 2310 psig and rising.

With the above conditions, the CVCS letdown line suddenly breaks just outside of the containment penetration and upstream of 1CV8152, Letdown Line Outboard Containment Isolation Valve.

Based upon the information above, what conditions would be expected following this event?

a. A leak will continue outside containment with NO valves closed to isolate it.

b. 1CV8149A, B & C, Orifice Isolation Valves, will immediately close on a Safety Injection Signal to isolate any leakage.

c. 1CV8149A, B & C, Orifice Isolation Valves, will immediately close on a Phase A Isolation Signal to isolate any leakage.

d. A leak would be present outside containment until 1CV8149A, B & C, Orifice Isolation Valves, fail CLOSED.

ANSWER
d.
REFERENCE
1BEP-0, Reactor Trip or Safety Injection
1BCA-1.2, LOCA Outside Containment
I1-CV-XL-01, Chemical and Volume Control System (Lesson Plan)
6E-1-4030CV21
6E-1-4030CV28
New
Higher
K/A Components, and Functions of Control and Safety Systems, Including Instrumentation, Signals, Interlocks, Failure Modes, and Automatic and Manual Features.
EXPLANATION
a, b, and c are all incorrect, because the orifice isolation valves upstream of the line break would be shut due to the Phase A signal generated by the SI signal.
d. is correct because the orifice isolation valves upstream of the line break would be shut due to the Phase A signal generated from the SI signal.

*QNUM 017
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A W/E11K201
QUESTION
During implementation of 1BCA-1.1, Loss of Emergency Coolant Recirculation, which of the following Unit 1 plant plant PARAMETERS/CONDITIONS are evaluated to determine the number of required Containment Spray pumps to operate?

1. RCFCs running in Accident Mode
2. Containment Pressure
3. RWST Level

a. 1, 2, and 3

b. 1 and 2 ONLY

c. 1 and 3 ONLY

d. 2 and 3 ONLY

ANSWER
a.
REFERENCE
1BCA-1.1, Loss of Emergency Coolant Recirculation
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the interrelations between the (Loss of Emergency Coolant Recirculation) and the following: Components, and functions of control and safety systems, including instrumentation, signals, interlocks, failure modes, and automatic and manual features.
EXPLANATION
a. is correct since the table in step 9 of 1BCA-1.1 determines number of CS pumps required based upon, RWST level, Containment Pressure, and RCFCs running in accident mode.

*QNUM 018
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A W/E05K102
QUESTION
The following Unit 1 plant conditions exist:

- Unit 1 has experienced a reactor trip
- A loss of all AF pumps has required an entry into 1BFR-H.1, Response to Loss of Secondary Heat Sink
- Containment pressure is 1 psig
- RCS temperature is 557 deg F

Which of the following plant conditions would require going to RCS Bleed and Feed immediately after tripping all RCPs?

a. Wide range SG levels are as follows: 1A 20%, 1B 30%, 1C 25%, 1D 30%.

b. PZR pressure is 2300 psig due to the loss of secondary heat sink.

c. RCS subcooling is unacceptable per the Iconic Display.

d. Loss of BOTH centrifugal charging pumps.

ANSWER
d.
REFERENCE
1BFR-H.1, Response to Loss of Secondary Heat Sink
FR-H.1, Background
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Normal, Abnormal, and Emergency Operating Procedures Associated with (Loss of Secondary Heat Sink)
EXPLANATION

d. Is correct, because, as per step 13 of 1BFR-H.1, if no charging pumps are available, it is required to skip to step 13 and secure RCPs. Then bleed and feed is established.

*QNUM 019
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000003A204
QUESTION
The following Unit 1 plant conditions exist:

- Unit 1 is at 40% power
- Control Bank D is at 180 steps
- Rods are in AUTO

The following then occurs:

- The control rods are observed to be stepping out.
- A control rod has dropped into the core by observation of the associated rod position indicator.

1) What parameters are used to determine that the control rod has dropped into the core?
2) How is control rod motion stopped during the event?

a. 1) Associated Rod Bottom light lit ONLY
   2) Rod motion is stopped by placing rod control in MANUAL

b. 1) Associated Rod Bottom light lit ONLY
   2) Rod motion is stopped by allowing Control Bank D to reach its auto rod stop setpoint of 223 steps

c. 1) Associated Rod Bottom light lit AND Tave initially dropping
   2) Rod motion is stopped by placing rod control in MANUAL

d. 1) Associated Rod Bottom light lit AND Tave initially dropping
   2) Rod motion is stopped by allowing Control Bank D to reach its auto rod stop setpoint of 223 steps

ANSWER
c.
REFERENCE
1BAR 1-10-D5, Bank D Rod Stop C-11
1BOA ROD-3, Dropped or Misaligned Rod
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to determine and interpret the following as they apply to the Dropped Control Rod: Rod motion stops due to dropped rod.
EXPLANATION

a. Is incorrect, because both Rod Bottom Lights and Tave dropping are required.
b. Is incorrect, because both Rod Bottom Lights and Tave dropping are required, and because the rod must be stopped in MANUAL as per 1BOA ROD-3, step 1.
c. Is correct.
d. Is incorrect, because the rod must be stopped in MANUAL as per 1BOA ROD-3, step 1.

*QNUM 020
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000005 2.1.7
QUESTION
The following plant conditions exist:

- Unit 1 is at 100% power.
- A control rod was determined to be misaligned by 25 steps from its Group Step counter position.
- It has been determined that it will take approximately 6 hours to realign the control rod to its Group Step counter position.

When the control rod realignment to its Group Step Counter position begins the crew must adhere to …

1. reactor power is limited to less than 50%.
2. reactor power must be no more than 75%.
3. the affected control rod must be moved into alignment at a rate of 3 steps/hour.

a. 1 ONLY

b. 2 ONLY

c. 1 and 3 ONLY

d. 2 and 3 ONLY

ANSWER
c.
REFERENCE
TS LCO 3.1.4, Rod Group Alignment Limits
1BOA ROD-3, Dropped or Misaligned Rod
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to evaluate plant performance and make operational judgments based on operating characteristics, reactor behavior, and instrument interpretation
EXPLANATION
a. Is correct and required by 1BOA ROD-3, step 10.
b. Is incorrect for the same reason as 1.
c. is the ONLY correct answer
d. Is correct and required by 1BOA ROD-3, step 19.

*QNUM 021
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000024K101
QUESTION
The following plant conditions exist:

- Unit 1 experienced a Reactor Trip
- 1 BEP ES-0.1, Reactor Trip Response, has been entered
- Tave is 550 deg F and dropping
- Two control rods do NOT have their rod bottom lights LIT

For the above plant conditions, what is the MINIMUM amount of emergency boration from the Boric Acid Storage Tank that is required at this time?

a. 1600 gallons

b. 2920 gallons

c. 3480 gallons

d. 5780 gallons

ANSWER
b.
REFERENCE
1BEP ES-0.1, Reactor Trip Response
1BOA PRI-2, Emergency Boration
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the operational implications of the following concepts as they apply to emergency boration: Relationship between boron addition and changes in T-ave.
EXPLANATION
a. Incorrect. Obtained by taking 280 gallons for the Tave drop (which is correct), but incorrectly taking only 1320 gallons for the rods that failed to fully insert.
b. Correct. Obtained by taking 280 gallons for the Tave drop (40 gal/ deg F X (557 deg F – 550 deg F), and 2640 gallons for the failure of 2 control rods to fully insert (1320 gallons/rod X 2 rods).
c. Incorrect. Obtained by incorrectly taking 840 gallons for the Tave drop (which would be the case if the RWST was used for emergency boration - - i.e., 120 gal/ deg F X (557 deg F – 550 deg F)), and correctly taking 2640 gallons for the failure of 2 control rods to fully insert
d. Incorrect. Obtained by taking 280 gallons for the Tave drop (which is correct), but incorrectly taking 5500 gallons (this is the amount of gallons from the RWST for one rod to fail to fully insert).

*QNUM 022
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000028K202
QUESTION
Select the set of conditions that would cause INDICATED Pressurizer Level to rise. (Assume leaks on variable and reference legs are NOT in the vapor space. Consider each condition separately.)

1. Leak on Variable Leg
2. Leak on Reference Leg
3. "Delta P" diaphragm ruptures
4. Containment temperature increases

a. 1, 3, and 4

b. 1, 2, and 3

c. 1, 2, and 4

d. 2, 3, and 4

ANSWER
d.
REFERENCE
Task 8, Respond to a PZR level control malfunction.
Higher
???NEW/USED
K/A Knowledge of the interrelations between the Pressurizer Level Control Malfunctions and the Sensors and detectors
EXPLANATION
a. Is incorrect because a leak in the variable leg (actual PZR level) would cause the delta P in the level instrument to rise causing indicated level to lower.
b. Is incorrect because a leak in the variable leg (actual PZR level) would cause the delta P in the level instrument to rise causing indicated level to lower.
c. Is incorrect because a leak in the variable leg (actual PZR level) would cause the delta P in the level instrument to rise causing indicated level to lower.
d. Is correct because all of these criteria cause the delta P in the level instrument to lower causing indicated level to rise.

*QNUM 023
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000032K301
QUESTION
The following plant condition exist:

Unit 1 is in MODE 2.

- Reactor thermal power is currently above P-6, but less than P-10.
- Calibration of N-31 is in progress with N-31 and N-35 Level Trip Switches in BYPASS.

Numerous alarms are subsequently received in the control room indicating a loss of Instrument Bus 112. Which of the following action(s) is (are) required IMMEDIATELY and why?

a. Suspend positive reactivity additions AND reduce power to less than P-6 to ensure protection against an uncontrolled rod withdrawal from a low power condition.

b. ONLY suspend positive reactivity additions to ensure protection against a continuous and uncontrolled RCCA bank rod withdrawal from a low power condition during startup, boron dilution and rod ejection events.

c. ONLY reduce power to less than P-6 to ensure protection against an uncontrolled rod withdrawal from a low power condition.

d. Verify the Reactor Trip Breakers are open to ensure protection against a continuous and uncontrolled RCCA bank rod withdrawal from a low power condition during startup, boron dilution and rod ejection events.

ANSWER
d.
REFERENCE
TS 3.3.1 and bases for Conditions F, G, H, I and Table 3.3.1-1; LP - Gamma-Metric Source and Intermediate Range Nuclear Instrumentation, Rev. 2, Section III.A.d. and e.; 1B0A INST-1, Nuclear Instrumentation Malfunction Unit 1, Rev. 104, Attachment C, SR Channel Failure.
New
Higher
K/A Knowledge of the reason(s) for Startup termination on source-range loss as applied to the Loss of Source Range Nuclear Instrumentation.
EXPLANATION
a. Correct actions and bases for loss of two IR channels.
b. Correct action and bases for loss of one SR channel.
c. Correct action and bases for loss of one IR channel.
d. LP System Description, bases and -Gamma-Metric Source and Intermediate Range Nuclear Instrumentation; 1B0A INST-1, Nuclear Instrumentation Malfunction Unit 1, Att. C, for loss of two SR channels.

*QNUM 024
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000033A103
QUESTION
Unit 1 is starting up with the following conditions:

- Reactor power is at 7%.
- Due to IR Channel N35 reading a full decade lower than IR Channel N36, Channel N35 has been placed in BYPASS.

While withdrawing rods in Control Bank D, IR Channel N36 fails low and the LOSS OF DETECTOR VOLT light on the N36 drawer is lit.

Which one of the following is a required response for this condition?

a. Immediately trip the reactor and follow required actions in 1BEP-0, Reactor Trip or Safety Injection.

b. Immediately reduce power to less than P-6.

c. Immediately stop control rod withdrawal and suspend any other positive reactivity additions.

d. Continue power ascension to greater than P-10.

ANSWER
c.
REFERENCE
Horse Notes, NI-3, Intermediate Range, Rev. 3; System Description, Gamma-Metric Source and Intermediate Range Nuclear Instrumentation, Rev. 2; 1B0A INST-1, Nuclear Instrumentation Malfunction Unit 1, Attachment B, IR Channel Failure.
Bank
Higher
K/A Ability to operate and / or monitor the Manual Restoration of Power as applied to the Loss of Intermediate Range Nuclear Instrumentation.
EXPLANATION
a. is plausible; would be appropriate action for loss of both SR detectors.
b. is plausible; required to reduce power below P-6 for loss of one IR Channel.
c. is correct
d. is plausible; alternative action to take for loss on one IR Channel.

*QNUM 025
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000061A203
QUESTION
During refuel operations on Unit 1, the following Area Radiation Monitors indications are received on the RM-11:

- 1AR11J, CONTAINMENT FH ICDT is flashing RED.
- 0AR55J, FUEL HANDLING BLDG FH ICDT is flashing YELLOW.

Which of the following Immediate Actions are to be taken? Verify automatic …

a. FHB Charcoal Booster Fan OVA04CA start.

b. Train A Containment Purge Isolation.

c. FHB Crane upward hoist movement inhibited.

d. Control Room Outside Air Intake A suction realignment to TB and Make-Up Fan start.

ANSWER
b.
REFERENCE
Horse Notes, AR/PR-1,FM-11,23,80, Rev. 1; Lesson Plan, Radiation Monitors, Rev. 5, Appendix B, Item 3 and Section II, System Description, Section H.2.; BOP AR/PR-11T1, Rad Monitor Interlock Function Table, Rev. 7; 1BAR RM11-4-1ARAAJ, Rev. 8; BAR RM11-4-0AR55J, Rev. 2.
New
Fundamental
K/A Ability to determine and interpret Setpoints for Alert and High Alarms as applied to the Area Radiation Monitoring (ARM) System.
EXPLANATION
a. is plausible; ARM interlock for High Alarm condition on 0AR55J
b. is correct
c. is plausible; ARM interlock for High Alarm condition on 0AR039
d. is plausible; PRM interlock for High Alarm condition on 0PR31J

zzz*QNUM 026
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 000037 2.2.22
QUESTION
The following plant conditions exist:

- SG 1A has just developed a tube leak.
- The leak rate in SG 1A has been determined to be 80 GPD.
- SG 1C has a previously determined tube leak of 40 GPD whose leak rate has been stable.
- SG 1D has a previously determined tube leak of 40 GPD whose leak rate has been stable.

Per 1 BOA SEC-8, Steam Generator Tube Leak, is a reactor shutdown required and what is the reason for the required shutdown, if required?

a. Yes, since SG 1A leak rate exceeds the Technical Specification limit.

b. Yes, since total SG leak rate on all 4 SGs exceeds the Technical Specification limit.

c. No, unless the SG 1A leak rate rises at a rate greater than allowable.

d. Yes, since SG 1A leak rate is above the procedural limit, though below the Technical Specification limit, for continued operation.

ANSWER
d.
REFERENCE
TS LCO 3.4.13 RCS Operational Leakage
1BOA SEC-8, Steam Generator Tube Leak
New
Higher
K/A Knowledge of Limiting Conditions for Operations and Safety Limits

EXPLANATION
a. Is incorrect, because TS LCO 3.4.13 limits for leakage from any ONE SG is 150 gallons per day.
b. Is incorrect, because TS LCO 3.4.13 limits are only based upon ONE SG as stated above.
c. Is incorrect, because the 80 GPD rate increase exceeds the procedural limit.
d. Is correct.

*QNUM	027
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	000051K301

QUESTION

Unit 1 is conducting a start-up, and with Reactor Power at 16%, had just closed the generator output breakers, when the following occurred:

- Main Condenser Vacuum lowered to the Low Vacuum Alarm setpoint
- 1BOA SEC-3, Loss of Condenser Vacuum was entered
- As vacuum reached 7.8 "HgA, the Reactor was tripped.
- 1 circulating water pump is supplying the Unit 1 condenser.

Which ONE of the following describes the current status of the steam dumps?

a. Steam dumps are armed.
b. Steam dumps are NOT armed because C-9 is NOT satisfied due to having only one circulating water pump running.
c. Steam dumps are NOT armed because C-9 is NOT satisfied due to low condenser vacuum.
d. Steam dumps are NOT armed because C-7 is NOT satisfied due to insufficient load rejection.

ANSWER

c.

REFERENCE

I1-OA-XL-38, BOA SEC-3, Loss of Condenser Vacuum
1BOA SEC-3, Loss of Condenser Vacuum
New
Fundamental
Proposed references to be provided to applicants during examination: Figure 1BOA SEC-3-1, Turbine Load –vs- Condenser Pressure

K/A Knowledge of the reasons for the following responses as they apply to the Loss of Condenser Vacuum: Loss of steam dump capability upon loss of condenser vacuum.

EXPLANATION

a. Is correct because C-9 is satisfied since condenser pressure has not exceeded 6.0 HGa and 1 circ pump is available, and C-7 is satisfied due to Steam Pressure Mode.
b. Is incorrect because C-9 is satisfied.
c. Is incorrect because C-9 is satisfied.
d. Is incorrect because C-7 is satisfied.

*QNUM 028
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 003K501
QUESTION
Unit 2 start up is in progress with Reactor Power at 16% and all systems normally aligned.

An electrical transient causes the 2A and 2C RCP Breakers to trip open.
2B and 2D RCPs remain running
The RCP Breaker Position Reactor Trip Circuit malfunctioned and NO Reactor trip occurred.

If NO operator action is taken, what will happen within 2 minutes?

The reactor will…

a. NOT automatically trip. RCS overpressure condition will NOT result.

b. NOT automatically trip. Excessive KW/ft condition will NOT result.

c. automatically trip. DNB condition will NOT result.

d. automatically trip. Loss of heat sink condition will result.

ANSWER
c.
REFERENCE
I1-RC-XL-02, Reactor Coolant Pump
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the operational implications of the following concepts as they apply to the RCPS: The relationship between the RCPS Flow Rate and the Nuclear Core operating parameters (quadrant power tilt, imbalance, DNB rate, local power density, difference in loop T-hot pressure.)
EXPLANATION
a. Is incorrect, because power exceeds the P-7 setpoint activating the loss of flow reactor trip.
b. Is incorrect for the same reason as A.
c. Is correct. The reactor will trip due to a Loss of Flow signal.
d. Is incorrect, because even though the RCP breaker position reactor trip did not actuate as required, this trip is only an anticipatory trip for the Loss of Flow trip. A loss of heat sink condition is not the result of a loss of RCS flow.

*QNUM 029
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 003K602
QUESTION
While following 1BCA-0.0, Loss of all AC Power, an operator locally closes 1CV8100, RCP Seal Water Return Containment Isolation Valve.

The reason 1CV8100 was closed was to…

a. prevent a failure of the RCP seals.

b. prevent steam formation in the CC lines of the Seal Water Heat Exchanger.

c. conserve RCS inventory.

d. protect the RCPs from seal and shaft damage that may occur when a charging pump is started as part of the recovery.

ANSWER
c.
REFERENCE
1BCA-0.0, Loss of all AC Power
ECA 0.0, Bases for Loss of all AC Power
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the effect of a loss or malfunction on the following will have on the RCPs: RCP seals and seal water supply.
EXPLANATION
a. Is incorrect, because the RCP is designed to operate with either seal injection or the thermal barrier heat exchanger in service.
b. Is incorrect, isolation of CC to the thermal barrier heat exchangers are done for this reason.
c. Is correct as outlined in ECA-0.0.
d. Is incorrect because this is the reason for isolating the seal injection line, not the seal return line.

*QNUM 030
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 004K509
QUESTION
BOP CV-1a(b), Startup of the CV System, states that transferring between the Loop A and Loop B Cold Leg charging return valves 1CV8146, RC CL Loop 2 Charging Water Inlet Valve, and 1CV8147, RC CL Loop 1 Charging Water Inlet Valve, should normally be done under Cold Shutdown conditions.

Failure to perform this evolution under Cold Shutdown conditions could result in:

a. over-pressurization of the cold leg return lines.

b. runout of the charging pumps.

c. thermal shock to the cold leg return lines.

d. an unplanned reactivity change.

ANSWER
c.
REFERENCE
BOP CV-1a(b), Startup of the CV System; Rev 21; Precaution 6
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the operational implications related to thermal shock (high component stress due to rapid temperature change) as they apply to CVCS.
EXPLANATION
a. Plausible due to difference in operating pressures between hot and cold plant conditions; CVCS designed for same operating pressure as Reactor Coolant system.
b. Plausible due to additional flow path if both lines placed in service at same time
c. Correct answer per BOP CV-1a(b)
d. Plausible if VCT boron concentration different from Reactor Coolant system and manipulation alters flow rate.

*QNUM 031
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 004K610
QUESTION
Unit 1 was operating at full power with all systems lined up in a normal full power configuration when a transient in the Chemical and Volume Control System occurred, resulting in the following indications:

- Letdown Relief Valve Temperature (TI125) 90 degrees F
- Regenerative Heat Exchanger Letdown Temperature (TI127) 300 degrees F
- Letdown Heat Exchanger Outlet Temperature (TI130) 150 degrees F
- Letdown Line Pressure (PI131) 230 psig
- Letdown Flow (FI132) 150 gpm

Which of the following malfunctions is the likely cause for the above indications?

a. Letdown Orifice Downsteam Relief Valve 1CV8117 failing open

b. Letdown Low Pressure Control Valve 1CV131 failing open

c. Low Pressure Control Valve Downstream Relief Valve 1CV8119 failing open

d. Letdown Demineralizer High Temperature Divert Valve1CV129 failing in the divert position

ANSWER
b.
REFERENCE
USAR Chapter 9; Table 9.3-5
CVCS Lesson Plan
New
Higher

K/A Knowledge of the effect, of a loss or malfunction, on the design minimum and maximum flow rates for letdown system.
EXPLANATION
a. would result in letdown flow decreasing
b. would cause relief valve 1CV-8119 to open(cycle), increased flow, temperature increase and diversion of flow to the VCT (from both the relief valve and divert valve)
c. CVCS parameters would be relatively unchanged from normal
d. CVCS parameters would be relatively unchanged from normal

*QNUM 032
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 005A102
QUESTION
Unit 2 has experienced a large break LOCA. The following conditions exist:

- 2A RH pump is out-of-service.
- 2B RH is in hot leg recirculation.
- A complete loss of instrument air has occurred.

Which one of the following describes the CORRECT response of the RHR system to the loss of instrument air with respect the following valves:

2RH607, 2B RHR HX Flow Control Valve
2RH619, 2B RHR HX Bypass Flow Control Valve
2RH611, 2B RH Pump Recirc Valve

a. 2RH607 fails Open, 2RH619 fails As-Is.

b. 2RH607 fails As-Is, 2RH611 fails Open.

c. 2RH607 fails Closed, 2RH611 fails Open.

d. 2RH607 fails Open, 2RH619 fails Closed.

ANSWER
d.
REFERENCE
RH-1, RHR Cooldown (Big Notes)
New
Fundamental
K/A Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the RHRS controls including: RHR flow rate.
EXPLANATION
a. Is incorrect because 2RH619 fails open.
b. Is incorrect because 2RH607 fails open and 2RH611 is a MOV.
c. Is incorrect because 2RH607 fails open and 2RH611 is a MOV.
d. Is correct, 2RH607 fails open while 2RH619 fails closed.

*QNUM	033
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	005A204

QUESTION

Following a large break LOCA and due to complications while implementing 1BEP ES-1.3, Transfer to Cold Leg Recirculation, the operators have shutdown both CS pumps due to RWST level. The CS pumps C/S's are in AFTER TRIP.

The CS pumps are directed to be restarted in step 9.d. RNO.

The 1A CS pump will NOT start from the MCB C/S in this step if…

a. 1CS019A, 1A CS Eductor Spray Add Valve, is closed.

b. 1SI8812A, 1A RH Pump RWST Suction Isolation Valve, is open.

c. 1CS007A, 1A CS Pump Discharge Header Isolation Valve, is closed.

d. 1SI8811A, 1A Containment Recirc Sump Outlet Isolation Valve, is closed.

ANSWER

d.

REFERENCE

I1-RH-XL-01, Residual Heat Removal System
2BEP ES-1.3, Transfer to Cold Leg Recirculation
RH-1, RHR Cooldown (Big Notes)
New
Higher
Proposed references to be provided to applicants during examination: None

K/A Ability to (a) predict the impacts of the following malfunctions or operations on the RHRS, and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: RHR valve malfunction.

EXPLANATION

a. Is incorrect because 1CS019A only interlocked with an auto start of the 1A CS pump.
b. Is incorrect because 1SI8812A is not interlocked with any start of the 1A CS pump.
c. Is incorrect because 1CS007 is only interlocked with a manual start of the 1A CS pump in the TEST position.
d. Is correct because 1SI8811A must be open to manually start the 1A CS pump.

*QNUM 034
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 006A201
QUESTION
Unit 1 was initially operating at 100% power when a safety injection occurred. The plant has entered into 1BEP-0, Reactor Trip or Safety Injection, to respond to the event. Present Unit 1 conditions are as follows:

- 1A Safety Injection Pump is Out-of-Service
- Containment pressure is 7.3 psig
- 1B SI Pump, 1A and 1B CV Pumps, and 1A and 1B RH Pumps are all running
- All RCPs are running
- RCS pressure is 1620 psig and slowly lowering
- Both PZR PORVs are closed
- RCS Temperature is 541°F and slowly lowering

The crew has learned that the thrust bearing temperature for the 1B SI pump is presently 208 degrees and rising; therefore, the 1B SI pump was stopped.

While at Step 25 of 1BEP-0, Reactor Trip or Safety Injection, which one of the following actions would be CORRECT in response to the event?

a. Stop RCPs. Stop dumping steam.

b. DO NOT stop RCPs. Establish a maximum cool down rate of 50F/Hr.

c. DO NOT stop RCPs. Stop dumping steam.

d. DO NOT stop RCPs. Continue to depressurize the RCS by dumping steam to the condenser from intact SGs.

ANSWER
c.
REFERENCE
1BEP-0, Reactor Trip or Safety Injection
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to predict the impact of high bearing temperature on the ECCS; and based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations.
EXPLANATION
a. Is incorrect because RCPs should not be stopped until RCS pressure is less than 1425 psig.
b. Is incorrect because maximum cool down rate would be 100°F/Hr.
c. Is correct because RCPs should not be stopped until RCS pressure is less than 1425 psig, and since RCS temperature is less than 557°F and decreasing, steam dumping should be stopped.
d. Is incorrect for the same reasons given in C, because RCS temperature decrease should be stopped, not increased.

*QNUM 035
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 006A302
QUESTION
Unit 2 is in MODE 4 with a plant cooldown in progress. The following plant conditions exist:

- RCS temperature is 300°F and slowly lowering due to the plant cooldown.
- 2A RH providing shutdown cooling.
- RCS pressure is 310 psig.
- LCO 3.4.12, Low Temperature Overpressure Protection (LTOP) System, is being met, and pressure relief capabilities for LTOP are met by the 2 PZR PORVs.

In these conditions, an inadvertent SI actuation occurs. With NO operator action, what would be the expected plant response? NOTE: Unit 2 LTOP PORV Setpoint Curve is provided.

a. One CV pump realigns to its ECCS lineup with the 2A RH suction relief valve being the first relief valve to lift.

b. BOTH CV pumps realign to their ECCS lineup causing pressure in the RCS to rise with the 2A RH suction relief valve being the first relief valve to lift.

c. One CV pump and BOTH SI pumps realign to their ECCS lineup causing pressure in the RCS to rise with the PORVs being the first relief valves to lift.

d. One CV pump realigns to its ECCS lineup with the PORVs being the first relief valves to lift.

ANSWER
a.
REFERENCE
LCO 3.4.12 and Bases, LTOP System
I1-RH-XL-01, Residual Heat Removal System
New
Higher
Proposed references to be provided to applicants during examination:
TRM LTOP PORV Setpoint Curve
K/A Ability to monitor automatic operation of the ECCS, including: pumps.
EXPLANATION

a. Is correct RH suction relief valve setpoint is 450 psig, 2RY455A lifts at 599 psig at this temperature, and 2RY456 lifts at 639 psig at this temperature.
b. Is incorrect because only one CV pump will be available for ECCS.
c. Is incorrect because SI pumps will not be available, and because the 2A RHR relief valve will lift first.
d. Is incorrect because the 2A RHR relief valve will lift first.

*QNUM 036
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 007A404
QUESTION
The following conditions exist for Unit 1:

- Reactor power - 100%
- RCS activity is elevated, but below Technical Specification (ITS) levels
- Pzr pressure - 2225 psig
- Pzr level - 60%

PORV 1RY456 opened due to a Pressurizer pressure instrument channel failure. The operator placed the 1RY456 control switch in the CLOSE position. When conditions stabilize:

- Reactor power - 100%
- Pzr pressure - 2228 psig
- Pzr level - 60%

How would the operator be able to tell if the PORV has fully seated?

a. Level change in RCDT.

b. Verify VCT level trends are normal.

c. Position lights for 1RY-456 showing CLOSE indication.

d. Lowering readings for containment radiation monitors 1AR011/012.

ANSWER
b.

EXPLANATION
a. Is incorrect because the PORV relief path flows to the PRT.
b. Is correct because the normal VCT level trend visibly shows RCP #2 seal leakoff. A deviation from this normal trend could indicate that the PORV is not seated (additional loss of RCS inventory).
c. Is incorrect because the light indication shows limit switch position only and not the actual seat position of the valve. The limit switch is a course indication of valve position.
d. Is incorrect because the activity level in the PZR remains unchanged following the cycling of a PZR PORV.

REFERENCE
Bank
Higher
K/A Ability to manually operate and/or monitor in the control room, the PZR vent valve.

*QNUM 037
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 008 2.1.20
QUESTION
A caution in BFR-C.1 states, “RH pumps should NOT be run longer than 2.4 HOURS without CC flow to the RH heat exchangers.” Which of the following statements best describes the basis for this caution?

a. Steam formation in the shell side of the heat exchangers could result in steam binding of the CC pumps and water hammer of CC piping upon restoration of flow.

b. If RCS pressure is above the shutoff head of the RH pumps, pump or motor failure may occur due to pump overheating or cavitation.

c. This limitation will minimize thermal stresses on the RH heat exchangers.

d. CC flow is necessary since it is possible that core cooling restoration will be provided via ECCS recirculation mode, and the sump water must be cooled prior to being delivered to the core or CV/SI pump suctions.

ANSWER
b.
REFERENCE
New
Higher
K/A Ability to execute procedure steps.
PROVIDE EXPLANATION

*QNUM 038
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 010K105
QUESTION
Unit 1 is operating with the following conditions:

- Reactor power is 100%.
- Pressurizer pressure is 2255 psig.
- Pressurizer pressure control is selected to 1PT455/456.

1PT 455 fails low. Assuming NO operator actions, what is the effect on RCS pressure, and the waste gas compressor?

Over the next 20 minutes, RCS pressure will cycle…

a. between 2335 and 2315 psig. The waste gas compressor will continue to remove gases from the PRT.

b. between 2335 and 2315 psig. The waste gas compressor will be isolated from the PRT.

c. at 2185 psig. The waste gas compressor will continue to remove gases from the PRT.

d. at 2185 psig. The waste gas compressor will trip on high suction pressure.

ANSWER
b.
REFERENCE
I1-RY-XL-01, Pressurizer (RY)
RY-2, PZR Pressure Control (Big Notes)
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the physical connections and/or cause-effect relationships between the PZR PCS and the following systems: PRTS
EXPLANATION
a. Is incorrect, because RY 469 will automatically close on a high pressure signal of 6 psig in the PRT.
b. Is correct.
c. Is incorrect, for the same reason as A., and because 1PT 455, as the controlling PT, will cause heaters to raise pressure until PORV 456 opens as 2335 psig.
d. Is incorrect for the same reasons as A. and C.

*QNUM 039
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 012K201
QUESTION
The Reactor is at 100% power. Which of the following will result in a Solid State Protection System Train B General Warning alarm?

1. The Ground Return Fuse in SSPS cabinet B is OPEN.
2. A loss of 120 VAC Instrument Bus 113.
3. A loss of a 15 VDC power supply to SSPS cabinet B.

a. 1 AND 3 ONLY.

b. 1, 2, AND 3.

c. 1 AND 2 ONLY.

d. 2 AND 3 ONLY.

ANSWER
a.
REFERENCE
I1-RP-XL-01, SSPS
BAR 1-4-B3, SOLID STATE PROT CAB GENERAL WARNING
SSPS-1, SSPS Block Diagram (Big Notes)
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of bus power supplies to the following: RPS channels, components, and interconnections.
EXPLANATION
a. Is Correct; number 2 is not correct, because SSPS cabinet B is supplied by Bus 114.

*QNUM 040
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 013K301
QUESTION
One hour ago, an SI occurred on Unit 2 due to a LOCA. The following conditions exist:

- Offsite Power is available and supplying power to the ESF buses.
- DGs are running but are NOT connected to the ESF buses.
- Containment pressure is 5.3 psig.
- Pressurizer level is 24% and slowly lowering.
- RCS pressure is 800 psig and lowering.
- ALL CV, SI, and RH pumps started and are running.
- The SI signal has been RESET.

With the above plant conditions, a Loss of Offsite Power occurs.

Assuming that NO operator actions are taken, which ONE of the following describes what will occur and the effect that this will have on core cooling?

Note: Figure 2BCA 1.1-1 is provided.

The DGs will re-energize their respective ESF buses,…

a. but ONLY the CV pumps will automatically restart causing INADEQUATE cooling of the reactor fuel.

b. but ONLY the CV pumps will automatically restart. However, the CV pumps will still be able to provide ADEQUATE cooling of the reactor fuel.

c. but the CV, SI, and RH pumps will NOT automatically restart causing INADEQUATE cooling of the reactor fuel.

d. and the CV, SI, and RH pumps WILL automatically restart resulting in only a short interruption in core cooling caused by sequencing times.

ANSWER
b.
REFERENCE
Figure 1BCA 1.1-1, Required ECCS Flow vs. Time from Trip
I1-DG-XL-01, Diesel Generator & Aux System
2BEP-01, Loss of Reactor or Secondary Coolant
New
Higher
Proposed references to be provided to applicants during examination: Figure 1BCA 1.1-1, Required ECCS Flow vs. Time from Trip
K/A Knowledge of the effect that a loss or malfunction of the ESFAS will have on the following: Fuel.
EXPLANATION
a. Is incorrect. Reference Figure 1BCA 1.1-1, Required ECCS Flow vs. Time from Trip.

b. Is correct.
c. Is incorrect, because the CV pump will reconnect.
d. Is incorrect, because an SI signal is no longer present and only the CV pumps will reconnect.

*QNUM 041
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 022K401
QUESTION
Unit 2 is operating at 100% power when the CNMT PEN CLG FLOW HIGH LOW annunciator alarms due to a loss of electrical power to valve 2CC053, Inside Containment Penetration Cooling Supply valve.

Which one of the following would be an effect of the loss of penetration cooling?

a. Inadequate cooling to the containment electrical penetrations could cause excessive heat and long term degradation to the penetration assemblies.

b. Inadequate cooling to the containment electrical penetrations could result in failure of the penetration assembly due to an electrical fault.

c. Piping penetrations with penetration cooling may overheat causing long term degradation to the containment structure.

d. Thermal expansion of piping with penetration cooling could cause cracking of the containment concrete.

ANSWER
c.
REFERENCE
BAR 2-2-D7, CNMT PEN CLG FLOW HIGH LOW
I1-PC-XL-01, Primary Containment
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of CCS design feature(s) and/or interlock(s) which provide for the following: Cooling of containment penetrations.
EXPLANATION
a. Is Incorrect because electrical penetrations are not serviced by the penetration cooling system.
b. Is incorrect for the same reason as a.
c. Is correct.
d. Is incorrect. Containment concrete is adversely affected long term; however, it is not due to piping expansion.

*QNUM 042
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 026A101
QUESTION
A Main Steamline break occurred in Unit 2 Containment 20 minutes ago. The plant is presently shutdown with the following conditions:

- Containment pressure is 12 psig.
- Containment Spray automatically actuated on containment pressure.

Based upon the above information, when can containment spray be secured?

a. Containment Spray must operate for at least 8 hours after initiation.

b. Containment Spray can be secured immediately.

c. Containment Spray can ONLY be terminated 8 hours after initiation AND after the Spray Additive Tank LO-2 level lights are LIT.

d. Containment Spray can ONLY be terminated after the Spray Additive Tank LO-2 level lights are LIT.

ANSWER
b.
REFERENCE
2BEP-1, Loss of Reactor or Secondary Coolant
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CSS controls including: Containment Pressure.
EXPLANATION
b. Is Correct
a, c, d are all incorrect because these limitations on securing containment spray are only applicable to large break LOCAs.

*QNUM 043
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 039K508
QUESTION
Reactor power is at 100% when the following events occur:

- The main turbine trips.
- The reactor does NOT automatically trip due to a failure of the Turbine Trip circuitry for the Reactor Trip System.

Assuming NO operator action, the reactor will still eventually automatically trip.

What Reactor Trip System Functions will initiate this reactor trip?

1. Overpower delta T.
2. LO-2 S/G Level.
3. Overtemperature delta T.
4. Pressurizer Pressure.

a. 1, 2, AND 3 ONLY.

b. 1 AND 4 ONLY.

c. 2 AND 3 ONLY.

d. 3 AND 4 ONLY.

ANSWER
d.
REFERENCE
Main Steam System, I1-MS-XL-01
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the operational implications of the following concepts as they apply to the MRSS: Effect of steam removal on reactivity.
EXPLANATION
d. Is correct, since, depending on the amount of reactivity feedback to the RCS, either an Overtemperature delta T OR High Presurizer Pressure condition could have caused the trip.

*QNUM 044
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 059A205
QUESTION
The following plant conditions exist:

- Generator Output	652 MW
- Feedwater Regulating Valves (FWRV)	AUTO
- 1A Motor Driven Main Feed Pump (MFP)	Running/Manual Operation
- 1B Turbine Driven MFP	Out of Service
- 1C Turbine Driven MFP	Running/AUTO Operation

Which of the following actions would occur if there is a suction line break to the 1A motor driven Main Feed Pump?

	1A MFP	1C MFP	FWRV
a.	continues running	speed lowers	throttle in open direction
b.	continues running	speed rises	throttle in close direction
c.	trips	speed rises	throttle in open direction
d.	trips	speed rises	throttle in close direction

ANSWER
c.
REFERENCE
I1-FW-XL-01, SG Water Level Control System
I1-CD-XL-01, Cond/FW System
Modified
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the MFW; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Rupture in MFW suction or discharge line
EXPLANATION
a. Is incorrect, because MFP speed will increase and 1A MFP will trip. The reactor does not have to be tripped unless above 700 MW.
b. Is incorrect, because the feed reg valve will open. The reactor does not have to be tripped unless above 700 MW.
c. Is correct.
d. Is incorrect, because 1A MFP will trip.

*QNUM 045
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 061K601
QUESTION

1AF004A, 1A AF Pump Discharge Valve, was closed for a surveillance run of the 1A Auxiliary Feedwater pump.

Before the pump was started, the air supply connection to the 1AF004A's air accumulator came loose and over the next 5 minutes SLOWLY depressurized the accumulator completely.

- Then, a Safety Injection occurs.

The 1A AF pump will…

a. NOT start automatically.
b. start automatically, and deliver designed flow to the Steam Generators
c. start automatically, and provide ONLY recirc flow.
d. start automatically, and deliver NO flow (deadhead).

ANSWER
b.
REFERENCE
1BEP-0, Reactor Trip or Safety Injection
I1-AF-XL-01, Auxiliary Feedwater System (AF)
060LSYS, Remote Shutdown Panel
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the effect of a loss or malfunction of the following will have on the AFW components: Controllers and positioners.
EXPLANATION
A is incorrect, the AF pump start is NOT interlocked with 1AF004A position.
B is correct because 1AF004A will fail open on slow air depressurization.
C is incorrect because 1AF004A will fail open on slow air depressurization.
D is incorrect because 1AF004A will fail open on slow air depressurization.

*QNUM 046
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 062A305
QUESTION
The following conditions exist in Unit 1:

- Reactor Power is 100%.
- A sudden voltage drop on the 4160 VAC buses occurred 30 seconds ago with voltage levels as follows:
- 3835 volts on Bus 141
- 3820 volts on Bus 142

20 seconds later, an SI signal is received. Which one of the following correctly describes the automatic response of the AC Power System?

a. The degraded voltage trip relays for both the 141 and 142 buses de-energize, sending a trip signal to all feed breakers to Buses 141 and 142. The bus undervoltage relays send a signal to start the 1A and 1B DGs.

b. The degraded voltage trip relay for the 142 bus de-energizes, sending a trip signal to all feed breakers to Bus 142. The 142 bus undervoltage relay sends a signal to start the 1B DG. Bus 141 remains energized from offsite power.

c. The degraded voltage trip relays for both the 141 and 142 buses de-energize, sending a signal directly to start the 1A and 1B DGs and to trip all feed breakers to Bus 141 and 142.

d. The degraded voltage trip relays for both the 141 and 142 buses de-energize after a 10 second time delay, sending a signal directly to start the 1A and 1B DGs and to trip all feed breakers to Bus 141 and 142.

ANSWER
a.
REFERENCE
I1-AP-XL-01, AC Electrical Power System
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to monitor automatic operation of the ac distribution system, including: Safety-Related indicators and controls.
EXPLANATION
a. Is correct.
b. Is incorrect, because feed breakers for both buses will be tripped.
c. Is incorrect, because the signal for DG start comes from the undervoltage relay not from the degraded voltage relay.
d. Is incorrect, because the signal for DG start comes from the undervoltage relay not from the degraded voltage relay, and there is no 10 second time delay.

*QNUM 047
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 063A402
QUESTION
Unit 1 is operating at 100% power with the following conditions:

- Bus 111 is being powered from the 111 battery charger.

The control room receives the 125V DC BATT CHGR 111 TROUBLE Annunciator due to a HIGH DC output voltage.

- Battery 111 has 2.12 Volts per cell in each of 58 cells.

Assuming that the crew takes NO actions, what would be the expected voltage at the output of the 111 Battery 1 minute after the annunciator alarms?

a. Greater than 140 VDC.

b. Greater than 130 VDC but less than or equal to 140 VDC.

c. Greater than or equal to125 VDC but less than or equal to 130 VDC.

d. Less than 125 VDC.

ANSWER
d.
REFERENCE
BAR 1-21-E8, 125V DC BATT CHGR 111 TROUBLE
I1-DC-XL-01, 125 VDC Power Systems (Training Plan)
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to manually operate and/or monitor in the control room: Battery discharge rate
EXPLANATION

a. Is Incorrect because the AC input breaker will TRIP for the 111 battery charger on high output voltage.
b. Is Incorrect because Voltage for 58 cells will be 123 VDC.
c. Is incorrect for the same reason as B.
d. Is correct, because the 111 battery has 58 cells which correspond to 123 VDC.

*QNUM 048
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 064 2.1.30
QUESTION
How would the RO know from his Main Control Room indications that the 1A DG is in “Local Control?”

1. The DG TROUBLE/FAIL TO START annunciator would be lit.
2. The white light located on the DG START/STOP switch would be lit.
3. The lights for the DG START/STOP switch would be dark.

   a. 1 ONLY.
   b. 2 ONLY.
   c. 1 AND 3 ONLY
   d. 1 AND 2 ONLY.

ANSWER
b.
REFERENCE
BAR …
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to locate and operate components, including local controls.
PROVIDE EXPLANATION

*QNUM 049
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 073K101
QUESTION
Unit 2 is operating with the following conditions:

- Reactor power is 100%.
- The Control Room Ventilation system is in a normal alignment with 0A VC train in operation.
- 0PR31J, Control Room Outside Air Train A Radiation Monitor is in OPERATE FAILURE.
- 0PR32J, Control Room Outside Air Train A Radiation Monitor is NORMAL.
- 0PR33J, Control Room Outside Air Train B Radiation Monitor is in OPERATE FAILURE.
- 0PR34J, Control Room Outside Air Train B Radiation Monitor is NORMAL.

With the above conditions, 0PR32J, Control Room Outside Air Train A Radiation Monitor, receives a spurious HIGH RADIATION signal.

Which ONE of the following correctly describes the response of the Control Room HVAC system?

Control Room HVAC makeup air was shifted to the…

a. Turbine Building due to the 0PR31J OPERATE FAILURE.

b. Turbine Building due to the 0PR32J HIGH RADIATION.

c. outside air due to the 0PR33J OPERATE FAILURE.

d. outside air due to the 0PR32J HIGH RADIATION.

ANSWER
b.
REFERENCE
BOP VC-1, Startup of Control Room HVAC
I1-AR-XL-01, Radiation Monitors
I1-VC-XL-01, Control Room HVAC System
New
Fundamental
K/A Knowledge of the physical connections and/or cause-effect relationships between the PRM system and the following systems: Those systems served by PRMs.
EXPLANATION
a. Is incorrect, because a single OPERATE FAILURE alarm does not cause Control Room HVAC to realign.
b. Is correct.
c. Is incorrect because outside air is the normal alignment and the coincidence to realign is 2/2 operate failure on the same train.
d. Is incorrect, because Control Room HVAC realigns to the Turbine Building.

*QNUM 050
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 076K204
QUESTION
Given the following plant conditions:
Bus 141 is deenergized because of a bus ground fault.

Unit 1 has a Safety Injection occur.

With NO operator action, Containment Chilled Water (WO) flow to the RCFC's is through…

a. Train A ONLY.

b. Train B ONLY.

c. BOTH Train A AND Train B.

d. NEITHER Train A NOR Train B.

ANSWER
d.
REFERENCE
BOP CC-14, Post LOCA Alignment of the CC System
I1-CC-XL-01, Component Cooling Water System
New
Higher
Proposed references to be provided to applicants during examination: M-66A, Sheet 1
K/A Knowledge of bus power supplies to the following: Reactor Building closed cooling water
EXPLANATION
Each train of chilled water to containment has isolation valves powered from each ESF division power supply, so a failure of one ESF bus will not prevent isolation of either train. 1WO056A to Train A RCFC's is shut, 1WO006B and 1WO056B to Train B RCFC's are shut.

*QNUM 051
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 076K303
QUESTION
Given the following plant conditions:
1A RCP seal injection flow failed to 0 gpm because its throttle valve clogged.
0A, 0B, 0C and 0D SX cooling tower fans all tripped.

Alarms RCP LOWER BRNG TEMP HIGH (1-7-C2) and RCP SEAL OUTLET TEMP HIGH (1-7-D3) for 1A RCP have just LIT at 184 degrees F.

BOTH Lower Bearing AND Seal Outlet Temperatures are rising at 1°F/minute.

In 30 minutes, which, if any, RCP trip setpoint will be exceeded?

a. Pump Lower Radial Bearing Temperature ONLY

b. Pump Seal Outlet Temperature ONLY

c. BOTH Pump Lower Radial Bearing Temperature AND Pump Seal Outlet Temperature

d. NEITHER Pump Lower Radial Bearing Temperature NOR Pump Seal Outlet Temperature

ANSWER
d.
REFERENCE
I1-SX-XL-01, Essential Service Water System
BOP CC-8, Isolation of CC between Units 1 and 2
BOP CC-14, Post LOCA Alignment of the CC System
1BOA PRI-7, Essential Service Water Malfunction
New
Higher
K/A Knowledge of the effect that a loss or malfunction of the SWS will have on the following: reactor building closed cooling water.
EXPLANATION
d. is correct. Alarms both come in at 184°F. In 30 minutes, both temperatures will be at 214°F. Bearing Temperature trip setpoint is 225°F, and the Seal Leakoff Temperature trip setpoint is 235°F, so neither is exceeded.

Effect 1. is incorrect because the CC water to the RCP will be secured due to the SI.
Effect 3. is incorrect because SX to the RCFC Containment Chiller will be secured due to the SI.
Effect 4. is incorrect because CC water to the Excess Letdown HX will be secured due to the SI.

*QNUM 052
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 078K303
QUESTION
The Service Air Compressors (SAC) are lined up as follows:

- 1A SAC Lead
- 1B SAC STDBY 2
- 2A SAC STDBY 1
- 2B SAC Lag

- Reactor power is 100%.

The Unit 2 SAC Air Receiver pressure dropped from 114 psig to 101 psig, and is now at 112 psig and rising.

With the above conditions, Bus 243 is lost. Concurrently, the 1A SAC trips due to overcurrent on the motor. Assuming NO operator actions and given the following events, which ones would be expected and in what sequence would they occur?

1. IA RCVR 1 PRESS LOW alarm.
2. SAC RCVR 1 PRESS LOW alarm.
3. 1B SAC starts.
4. 2A SAC starts.

a. 2, and then 3 ONLY.

b. 2, and then 4.

c. 2, 3, and then 1.

d. 1, 2, and then 3.

ANSWER
a.
REFERENCE
SA-2, SAC and IA Dryer (Big Notes)
I1-SA-XL-01, Service Air and Instrument Air
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of effect that a loss or malfunction of the IAS will have on the following: Cross-tied units
EXPLANATION
a. is the correct answer. The IA RCVR 1 PRESS LOW alarm comes in at 108 psig. The 1B SAC will start at 104 psig. Since the stem of the question states that pressure was initially 114 psig, only the 1A SAC could have been running. Since pressure recovered at 101 psig, only the 2B SAC could have been running. Since only 2 air compressors were necessary to raise pressure, the 2 available SACs would recover pressure prior to the IA RCVR 1 PRESS LOW alarm.

*QNUM 053
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 078K401
QUESTION
The following conditions exist on Unit 1:

- DC Bus 111 was lost due to a ground fault.

The Operations crew is performing 1BOA ELEC-1, Loss of DC Bus, and are presently attempting to restore instrument air to the containment by opening 1IA065, IA CNMT outside isolation valve, and 1IA066, IA CNMT inside isolation valve.

Which one of the following is the procedurally required method for restoring instrument air to the containment?

a. Open 1IA065 manually from the Main Control Room; locally unlock and open 1IA066.

b. Open 1IA065 manually from the Main Control Room; open 1IA066 from the Main Control Room.

c. Locally unlock and open 1IA065; open 1IA066 from the Main Control Room.

d. Locally unlock and open 1IA065; locally unlock and open 1IA066.

ANSWER
c.
REFERENCE
SA-2, SAC and IA Dryer (Big Notes)
I1-SA-XL-01, Service Air and Instrument Air
1BOA ELEC-1, Loss of DC Bus
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of IAS design feature(s) and/or interlock(s) which provide for the following: Manual/automatic transfers of control.
EXPLANATION
a. Is incorrect, but it would be correct if DC Bus 112 were lost.
b. Is incorrect, because 1IA065 must be locally opened.
c. Is correct.
d. Is incorrect, because 1IA066 should be manually opened in the Main Control Room.

*QNUM 054
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 103K404
QUESTION
The following conditions exist in Unit 1:

- The Reactor is shut down in Mode 3.
- Containment pressure is 0.7 psig.
- You have made an emergency containment entry to investigate a steam leak, and are presently attempting to exit the containment through the personnel airlock doors.

While attempting to exit, you discover that the interior personnel airlock door will NOT open. Five minutes after mechanically opening the interior equalizing valve, it is discovered that pressure has still NOT equalized across the interior door.

Which of the following could be the reason(s) for this condition? (Consider each condition separately)

1. The exterior equalizing valve is closed.
2. The exterior equalizing valve is open.
3. Containment pressure is too high to allow the inner airlock door to open.

a. 1 AND 3 ONLY.

b. 2 AND 3 ONLY.

c. 1 ONLY.

d. 2 ONLY.

ANSWER
d.
REFERENCE
BAP 1450-8, Primary Containment Equipment/Emergency Hatch Personnel Airlock Doors Operation
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of containment system design feature(s) and/or interlock(s) which provide for the following: Personnel access hatch and emergency access hatch.
EXPLANATION
d. is correct, because with the interior equalizing valve open, either the equalizing valve or the exterior door (or both) is open (reference BAP 1450-8, step 4.6.4.3).

*QNUM 055
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 103A101
QUESTION
Unit 2 is operating at power with the following conditions;

-2A and 2C RCFCs are running in HIGH speed
-2B and 2D RCFCs are in STANDBY

Subsequently, a lightning strike results in a sudden pressure trip of SAT 242-2. Which of the following describes the status of the RCFCs one minute after the lightning strike?

a. All four RCFCs will be stopped.

b. The 2A and 2C RCFCs will be running in LOW speed, and the 2B and 2D RCFCs will be in STANDBY.

c. All four RCFCs will be running in LOW speed.

d. The 2A and 2C RCFCs will be running in HIGH speed, and the 2B and 2D RCFCs will be in STANDBY.

ANSWER
d.
REFERENCE
I1-VP-XL-01, Containment Ventilation and Purge System
VP-3, Containment Cooling (Big Notes)
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the containment system controls including: Containment pressure, temperature, and humidity.
EXPLANATION
d. Is correct because the RCFC breakers are 480 volt unit substation breakers that remain connected to their respective ESF busses. A sudden pressure trip of SAT 242-2 also results in a trip of SAT242-1. Both 241 and 242 deenergize and are reenergized by the diesel generators. Only a control switch manipulation starts the fan in high speed and only an SI signal starts the fan in low speed. The 480 volt breaker positions remain unchanged and therefore the same fan conditions are experienced both before and after the event.

*QNUM 056
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 001A105
QUESTION
The following conditions exist in Unit 1 following a refueling outage:

- Reactor power is 12% with the turbine offline.
- The Makeup Mode Selector Switch is in AUTO position.
- VCT level is 32%.

With the conditions above, a calculation error resulted in the Boric Acid Flow Control Potentiometer to be set for 10 ppm boron concentration INSTEAD of 1000 ppm boron concentration.

Assuming that NO operator actions are taken, what would be the eventual result?

a. Rods would step in automatically to compensate for the rise in Tave.

b. Rods would step out automatically to compensate for the lowering of Tave.

c. Reactor power would lower with no automatic rod compensation.

d. Reactor power would rise with no automatic rod compensation.

ANSWER
d.
REFERENCE
I1-CV-XL-02, Reactor Makeup Control System
I1-RD-XL-01, Rod Control System
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to predict and/or monitor changes in parameters (to prevent exceeding design limits) associated with operating the CRDS controls including: Effect on T-ave of dilution without rod motion compensation.
EXPLANATION
a. Is incorrect, because the Rod Control System is not in automatic .
b. Is incorrect for the same reason as A.
c. Is incorrect, because reactor power would increase due to dilution.
d. Is correct.

*QNUM 057
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 002A201
QUESTION
The following conditions exist in Unit 1:

- The plant experienced a large break LOCA 30 minutes ago.
- The operators have taken appropriate actions in accordance with 1BEP-0, Reactor Trip or Safety Injection, and 1BEP-1, Loss of Reactor or Secondary Coolant.
- RCS pressure is stable at 300 psig.
- ECCS pumps are operating in the cold leg recirculation mode.

In this mode, core decay heat is being removed PRIMARILY by:

a. The condensation of reflux boiling in the S/G's.

b. Heat transfer between the RCS and the S/G's due to forced circulation flow.

c. The injection of water from the recirculation sump and the removal of steam/water out from the break.

d. Heat transfer between the RCS and the S/G's due to Natural Circulation flow.

ANSWER
c.
REFERENCE

Bank
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the RCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of coolant inventory.
EXPLANATION
c. Is correct, because for the conditions in place, with a large break LOCA, heat would still be removed "primarily" by ECCS injection.

*QNUM 058
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 011A303
QUESTION
The following conditions exist in Unit 2:

- Reactor power is 100%.
- Pressurizer level is 60%.

With the conditions above and NO operator action, Level Transmitter, 2LT460, fails low. How does this event affect letdown and 2CV-121, Charging Flow Control Valve?

a. Letdown isolates, 2CV-121 throttles in the CLOSED direction.

b. Letdown isolates, 2CV-121 throttles in the OPEN direction.

c. Letdown does NOT isolate, 2CV-121 throttles in the CLOSED direction.

d. Letdown does NOT isolate, 2CV-121 throttles in the OPEN direction.

ANSWER
a.
REFERENCE
I1-RY-XL-01, Pressurizer (RY)
RY-3, PZR Level Control (Big Notes)
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to monitor automatic operation of the PZR LCS, including: Charging and letdown.
EXPLANATION

a. Is correct.
b. Is incorrect, because 2LT459 will detect a rising level in the PZR due to the letdown isolation and send a signal to close 2CV-121.
c. Is incorrect, because letdown will isolate.
d. Is incorrect for the same reasons as C. and D.

*QNUM 059
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 014A401
QUESTION
An extra NSO has just reported to the Unit Supervisor that while performing a task in the back of 1PM05J he repositioned the DRPI Accuracy Mode Selector Switch from the **A+B position** to the **A Only** position.

As a result …

a. ALL General Warning LED's will be flashing and the DRPI Rod Control Urgent Failure Alarm will actuate

b. DRPI will be in Half Accuracy Mode

c. Central Control Failure lights will be illuminated

d. DRPI will be in normal accuracy

ANSWER
b.
REFERENCE
DRPI Lesson Plan
Figure 29-8 DRPI Display Panel Controls
Figure 29-9 DRPI Accuracy
Big Note RD-6, Digital Rod Position Indication
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to manually operate and / or monitor in the control room:
PROVIDE EXPLANATION

a. Is incorrect because an Urgent Failure alarm will not actuate with only 1 channel disabled.
b. Is correct because with channel A only, half accuracy of +4, -10 steps will occur for DRPI.
c. Is incorrect because this is what would happen if channel B data were lost while selected to DRPI A only.
d. Is incorrect because DRPI will be in Half Accuracy Mode.

*QNUM 060
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 015 2.2.24
QUESTION
The following plant conditions exist:

- Power ascension is in progress
- Nuclear Instrumentation channels indicate the following

N-41	9%
N-42	11%
N-43	11%
N-44	8%

- Overpower Trip Low Range light is lit on Power Range Drawer N-42

Based on the above conditions what actions are required to be taken to address these indications?

a. Lower reactor power to below P-10

b. Enter BEP-0, Response to a Reactor Trip or Safety Injection

c. Manually Block the Power Range High Flux Low Setpoint Reactor Trip. No Technical Specification entry is required.

d. No actions are required since the Power Range High Flux Low Setpoint Reactor Trip is Automatically Blocked above 10% Reactor Power.

ANSWER
c.
REFERENCE
Technical Specification 3.3.1, RTS Instrumentation
Big Note NI-1 & NI-2 Power Range Detector
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to recognize system parameters that are entry-level conditions for Technical Specifications.
PROVIDE EXPLANATION
a. Is not correct because tech spec 3.3.1 for the high flux low power setpoint doesn't apply > P-10 (10%)
b. Is not correct because the reactor did not trip.
c. Is correct because the tech spec doesn't apply > P-10 and the High Flux Low Setpoint must be manually blocked.
d. Is not correct because the High Flux Low Setpoint must be manually blocked.

*QNUM 061
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 016K108
QUESTION
The following plant conditions exist on Unit 1 during a normal plant shutdown:

- RCS pressure is 395 psig
- RCS temperature is 345°F
- 1RY455A and 1RY456 are in Low Temperature Overpressure Protection mode.

Pressurizer PORV 1RY456 has just opened. Which one of the following caused 1RY456 to open?

A) 1PT-403 has failed high

B) 1PT-405 has failed high

C) 1PT-406 has failed high

D) 1PT-407 has failed high

ANSWER
c.
REFERENCE
Big Note RY-1 Pressurizer
TS 3.4.12
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the physical connections and/or cause-effect relationships between the NNIS and the following system – Pressurizer Pressure Control System.
EXPLANATION
a. Is not correct because 1PT-403 does not an input to PZR PORV 1RY456
b. Is not correct because 1PT-405 does not an input to PZR PORV 1RY456
c. Is correct because 1PT-406 inputs into PZR PORV 1RY456
d. Is not correct because 1PT-407 does not an input to PZR PORV 1RY456

*QNUM 062
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 055K301
QUESTION

Unit 2 is operating at steady state 100% power when 1MS053A, Steam Pressure Regulating valve to the only operating Steam Jet Air Ejector fails SHUT.

The operating crew enters the appropriate abnormal operating procedure and initiates actions to start the standby Steam Jet Air Ejector.

How does the failure of the operating Steam Jet Air Ejector **INITIALLY** affect Unit 2 assuming that the MW feedback loop is in "MW OUT" and first stage impulse pressure feedback loop is "IMP IN"?

	Condenser pressure	Electrical Megawatts
a.	RISES	LOWERS
b.	CONSTANT	LOWERS
c.	RISES	CONSTANT
d.	CONSTANT	CONSTANT

ANSWER
a.
REFERENCE
Lesson Plan Main Turbine Controls and Protection
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the effect that a loss or malfunction of the CARS will have on the Main Condenser.
EXPLANATION

a. Is correct. With loss of air removal, condenser pressure rises and with IMP IN holding turbine 1st stage pressure constant, electrical output will lower (DP across the turbine lowers).
b. Condenser Pressure will not remain constant due to no longer removing non-condensable gasses with the SJAEs.
c. Electrical Megawatts will not remain constant since turbine work is decreasing due to the increase in condenser pressure.
d. Condenser Pressure will not remain constant due to no longer removing non-condensable gasses with the SJAEs.

*QNUM	063
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	068K401

QUESTION

What design features, interlocks and administrative controls are in place to prevent an inadvertent discharge from the station liquid release tanks?

0WX353, Release Tank Pumps Discharge Valve and 0WX896, Release Tank Pumps Discharge Low Flow Valve…

a. have administrative clearance orders hung on their control switches.

b. are isolated until the normally closed manual upstream isolation valves 0WX352 and 0WX895 are locally opened.

c. are key locked valves.

d. have robust operational barriers covering the control switches (Plexiglas covers).

ANSWER

c.

REFERENCE

BCP 400-TWX01

New

Fundamental

K/A Safety and environmental precautions for handling hot, acidic, and radioactive liquids.

EXPLANATION

a. Not correct because and admin C/O is not hung on 0WX353 and 0WX896.
b. Not correct because manual upstream isolation valves are not closed upstream of the discharge valves.
c. Correct, 0WX353 and 0WX896 are key locked valves.
d. Not correct because these valve's control switches do not have Plexiglas covers over them.

*QNUM 064
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 071K504
QUESTION
Maintenance has just been completed on 0GW01TC, Gas Decay Tank. Which of the following conditions would require the 0C GDT to be purged with Nitrogen?

a. A Hydrogen concentration approaching 2% in the presence of Oxygen.

b. A Hydrogen concentration approaching 4% in the presence of Nitrogen.

c. A Oxygen concentration approaching 0.5% in the presence of Hydrogen.

d. A Oxygen concentration approaching 2% in the presence of Hydrogen.

ANSWER
d.
REFERENCE
BOP GW-5
New
Fundamental
K/A Relationship of Hydrogen / Oxygen concentrations to flammability
EXPLANATION
a. Not correct, because hydrogen conc. needs to be approaching 4% to purge.
b. Not correct, because 4% hydrogen is ok in the presence of Nitrogen.
c. Not correct, because oxygen conc needs to be approaching 2% to purge
d. Correct per limitation and action E.1 of BOP GW-5.

*QNUM 065
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 086K604
QUESTION
Given the following information:

- A fire has started in the 1A DG Room.
- The thermal detectors in the 1A DG Room are out-of-service for maintenance.
- The fire has activated one of the fire detectors in the room.

Which detector(s) would have activated, and what would be the result of the activation?

a. Ionization Detector(s) resulting in the actuation of the deluge system.

b. Ionization Detector(s) resulting in the actuation of the CO2 suppression system.

c. Ultraviolet Detector(s) resulting ONLY in an alarm on Fire Detection Panel 1PM09J.

d. Ultraviolet Detector(s) resulting in the actuation of the CO2 suppression system.

ANSWER
c.
REFERENCE
I1-FP-XL-01, Fire Protection System
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the effect of a loss or malfunction on the following will have on the Fire Protection System: Fire, smoke, and heat detectors
EXPLANATION
a. Is incorrect, because there is no deluge system in the 1A DG Room.
b. Is incorrect, because the thermal detectors activate the CO2 suppression system.
c. Is correct.
d. Is incorrect, because the thermal detectors activate the CO2 suppression system.

*QNUM	066
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	2.1.24

QUESTION

Which of the following prints would show the Flow Control Loop for the 0VC03CA Make-Up Fan?

a. 3040 series of prints

b. 3041 series of prints

c. 4030 series of prints

d. 4031 series of prints

ANSWER
d.
REFERENCE
0-4031VC04
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to obtain and interpret Station Electrical and Mechanical Drawings
EXPLANATION

*QNUM 067
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.1.32
QUESTION
The Unit Supervisor requests that you open the 2FW009A-D Feedwater Isolation valves.

Before you open these valves which combination of permissives must be met?

a. 2/3 FW temperature RTD’s at SG inlet is > 255°F for > 8 minutes
Feedwater flow through 2FW046 between 30 – 120 gpm for > 8 minutes
Total Feedwater flow supplied to specific SG > 1130 gpm

b. 2/3 FW temperature RTD’s at 2/3 locations is > 255°F for > 8 minutes
Feedwater flow through 2FW046 between 130 – 220 gpm
Total Feedwater flow supplied to specific SG > 1310 gpm

c. 2/3 FW temperature RTD’s at 3/3 locations is > 255°F for > 8 minutes
Feedwater flow through 2FW046 between 80 – 120 gpm for > 8 minutes
Total Feedwater flow supplied to specific SG > 1130 gpm

d. 2/3 FW temperature RTD’s at 3/3 locations is > 255°F for > 8 minutes
Feedwater flow through 2FW046 between 130 – 220 gpm for > 8 minutes
Total Feedwater flow supplied to specific SG > 1310 gpm

ANSWER
d.
REFERENCE
Lesson Plan 25r05, Condensate and Feedwater
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to explain and Apply all System Limits and Precautions.
EXPLANATION
d. Correct. To open 2FW009A-D the Purge Permissive and Flow Permissive must be met along with > 600psig and > 5% NR level in the associated SG.

*QNUM 068
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.1.33
QUESTION
Unit 1 is at 100% power.
You are the Unit 1 Assist NSO.
The following plant conditions exist:

- Unit 1 is at 100%.
- 2 gpm seat leakage from the Letdown Header Relief Valve to the PRT
- ACB 1411, Bus 141 to 143 Crosstie breaker, control power fuses are blown
- Pressurizer level is 70%

Based on the conditions listed above what Technical Specification action requirement do you need to inform the Unit Supervisor to enter?

a. Tech Spec 3.4.9, Pressurizer

b. Tech Spec 3.8.1, AC Sources-Operating

c. Tech Spec 3.4.13, RCS Operational LEAKAGE

d. Tech Spec 3.8.9, Distribution Systems-Operating

ANSWER
a.
REFERENCE

New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to recognize indications for System Operating Parameters which are Entry-Level Conditions for Technical Specifications.
EXPLANATION
a. is correct because redundant non-ESF power supplies must be available for the Pzr heaters. Pzr level of 70% is not a reason to enter this LCOAR, but is the high level setpoint and included as a reason to prevent this being an obvious answer.
b. & d. are incorrect, but are included as likely distractors for the ACB control power failure.
c. is incorrect, identified leakage up to 10 gpm is allowed.

*QNUM 069
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.2.1
QUESTION
Unit 1 is performing 1BGP 100-2, Plant Startup. The Reactor Operator is withdrawing Shutdown Bank A.

What will be the indicated rod speed and what will be the actual rod speed as Shutdown Bank A steps out?

	Indicated speed	Actual speed
a.	0 steps/minute	72 steps/minute
b.	48 steps/minute	48 steps/minute
c.	8 steps/minute	64 steps/minute
d.	64 steps/minute	64 steps/minute

ANSWER
d.
REFERENCE
1BGP 100-2, Plant Startup
Big Note RD-2, Reactor Control Unit
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to perform Pre-Startup Procedures for the facility, including operating those controls associated with plant equipment that could affect Reactivity.
EXPLANATION
a. Not Correct. Indicated speed will be 64 spm and actual will be 64 spm
b. Not Correct. Indicated speed will be 64 spm and actual will be 64 spm
c. Not Correct. Indicated speed will be 64 spm and actual will be 64 spm
d. Correct.

*QNUM 070
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.2.33
QUESTION
Assuming that the Rod Control Bank fully withdrawn position is 231 steps with bank overlap at 116 steps,

1. Where will the Bank Overlap Unit Control Rod Bank C **Start Position** thumbwheel setpoint be set, and
2. What is the reason for Control Rod Bank Overlap?

	START POSITION	REASON
a.	225	Increases differential rod worth. Maintains a more uniform flux distribution.
b.	225	Increases Shutdown Margin. Maintains a uniform differential rod worth.
c.	232	Maintains a more uniform differential rod worth. Maintains a more uniform flux distribution.
d.	232	Maintains a more uniform flux distribution. Reduces potential Axial Flux Offset Power Peaks

ANSWER
c.
REFERENCE
Rod Control Lesson Plan
BOP RD-7
Big Note RD-2 & RD-3
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of Control Rod Programming.
EXPLANATION
a. Not Correct. Rod Bank Overlap does not increase differential rod worth.
b. Not Correct. Rod Bank Overlap does not increase Shutdown Margin.
c. Correct.
d. Not Correct. 236 is not correct and Rod Bank Overlap does not reduce Axial Flux Offset.

*QNUM 071
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.3.10
QUESTION
A Steam Generator Tube Rupture has occurred on Unit 1.

- 1BEP-3, Steam Generator Tube Rupture procedure is being performed.
- Ruptured Steam Generator pressure and RCS pressure has been equalized to minimize RCS leakage.
- The Unit Supervisor has just ordered the performance of BOP MS-11, Operation with Steam Generator Tube Leakage.

The performance of BOP MS-11 will…

a. help chemistry identify contaminated secondary systems.

b. minimize secondary system contamination.

c. re-align radwaste demineralizers in anticipation of liquid radwaste processing.

d. align supplemental radiation monitors on the MS system.

ANSWER
b.
REFERENCE
1BEP-3, Steam Generator Tube Rupture
BOP MS-11, Operation with Steam Generator Tube Leakage
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to Perform Procedures to Reduce Excessive Levels of Radiation and Guard Against Personnel Exposure.
PROVIDE EXPLANATION

a. Not Correct. This procedure will not help chemistry identify contaminated systems.
b. Correct. This procedure will minimize the amount of contamination in the secondary system by re-aligning systems away from the contamination. Other systems are separated from the affected unit to keep the contaminated water on that unit only.
c. Not Correct. The only thing MS-11 does in anticipation of liquid radwaste processing is to verify Release tanks and Blowdown Monitor tanks are empty. This procedure does nothing to radwaste demins.
d. Not Correct. No additional radiation monitors are installed using this procedure.

*QNUM 072
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.3.11
QUESTION
A Unit 2 Containment Purge has been initiated in accordance with BOP VQ-5, Containment Purge System Operation. 2VQ05C, Unit 2 Containment Mini Purge Exhaust Fan has just automatically tripped.

Which of the following was responsible for causing 2VQ05C to trip?

a. Containment Spray was Manually Actuated

b. 2AR021J Containment Area Radiation Monitor went into High Alarm

c. 2PR01J Containment Purge Effluent Radiation Monitor went into High Alarm

d. Containment Exhaust Smoke Detector 2XY-VQ102 went into High Alarm

ANSWER
a.
REFERENCE
2-4030VQ013 Electrical Print
2-4030VQ01 Electrical Print
Big Note VP-2, Cnmt Purge
New
Higher

Proposed references to be provided to applicants during examination: None
K/A Ability to Control Radiation Releases.
EXPLANATION
a. Correct. A Manual Containment Spray initiation will close Cnmt Vent Isolation Valves causing 2VQ005C to trip.
b. Not Correct. 2AR021J Cnmt Area Radiation Monitor will not actuate a Cnmt Vent Isolation signal and thus will not result in a trip of 2VQ005C.
c. Not Correct. 2PR01J Containment Prge Effluent Rad Monitor will not cause a Cnmt Vent Isolation signal and thus will not result in a trip of 2VQ005C.
d. Not Correct. 2XY-VQ102 is the Cnmt Purge Supply ionization detector and does not cause a trip of 2VQ005C.

*QNUM	073
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	B
K/A	2.4.12

QUESTION

1BEP-0, Reactor Trip or Safety Injection Unit 1, has just been entered for indications of a Large Break LOCA. Which of the following personnel may be designated to monitor the Status Trees and when is the monitoring required to be initiated?

	Personnel	Start Monitoring
a.	Duty STA	Immediately upon entry into 1BEP-0
b.	Non-Licensed Operator trained on Status trees	When directed in 1BEP-0
c.	Unit 2 SRO	Immediately after step 4 of 1BEP-0
d.	Unit 2 Assist RO	Immediately after transition out of BEP-0

ANSWER
d.
REFERENCE
BAP 1310-10, revision 10, HU-AA-104-101, Procedure Use and Adherence, Byron Addendum

New
Higher
K/A Ability to execute procedure steps.

PROVIDE EXPLANATION
a. Is incorrect because monitoring of status trees is required to be started after transitioning out of 1BEP-0 or when directed within 1BEP-0.
b. Is incorrect because status trees are required to be monitored by a licensed individual.
c. Is incorrect because monitoring of status trees is required to be started after transitioning out of 1BEP-0 or when directed within 1BEP-0.
d. Is correct.

*QNUM 074
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.4.24
QUESTION
Unit 1 has experienced a Steam Generator Tube Rupture on the 1C Steam Generator. The Turbine had to be locally tripped but all other systems functioned normally, and the following plant conditions are observed:

- 1A SG NR Level is 4% and rising
- 1B SG NR Level is 3% and rising
- 1C SG NR Level is 8% and rising
- 1D SG NR Level is 3% and rising
- Pressurizer Pressure is 1735 psig and lowering
- Pressurizer Level is 18% and lowering
- 1AF013A-H are ALL OPEN.

The Crew has performed the actions of 1BEP-0, Reactor Trip or Safety Injection and has transitioned into 1BEP-3, Steam Generator Tube Rupture. The Crew is at step 4 to check ruptured Steam Generator revel. Based on the above indications the crew should…

a. Isolate ALL flow TO and FROM the 1C Steam Generator.

b. Isolate flow FROM the 1C Steam Generator Blowdown valves and 1C MSIV & 1C MSIV Bypass valves, but allow AF flow to continue to ALL Steam Generators.

c. Isolate all AF flow TO the 1C Steam Generator but do NOT isolate flow FROM 1C Steam Generator Blowdown valves, MSIV, and PORV.

d. Do NOT isolate any flow path on the 1C Steam Generator until AFTER a cooldown is initiated.

ANSWER
b.
REFERENCE
1BEP-3, Steam Geneator Tube Rupture
WOG Background Document for E-3 Steam Generator Tube Rupture
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of Loss of Cooling Water Procedures
EXPLANATION

a. Not Correct. NR SG level in ruptured SG is less than 10% so AF flow should not be isolated.
b. Correct. NR SG level in ruptured SG is less than 10% so AF flow is allowed to continue so that thermal stratification within the SG will be established.
c. Not Correct. The Crew must isolate flow paths out of SG and allow AF flow to continue to ruptured SG.
d. Not Correct. Need to isolate ruptured SG prior to cooldown initiation.

*QNUM 075
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level B
K/A 2.4.39
QUESTION
The following time line of events occurred on Unit 1:

- 1000 PZR Level started lowering
- 1001 SJAE / GS Exhauster Radiation Monitor 1PR27J went into high alarm
- 1005 Unit Supervisor orders a Reactor Trip and Manual Safety Injection
- 1010 Shift Manager Classified the Event as an Alert (FA1) due to 1A SGTR
- 1015 Crew enters 1BEP-3, Steam Generator Tube Rupture

In order to meet the notification requirements for NARS, the INITIAL notification to the State and Local Agencies must be made NO LATER THAN….

a. 1015

b. 1016

c. 1020

d. 1025

ANSWER
d.
REFERENCE
EP-AA-114, Notifications
Bank
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the RO's Responsibilities in Emergency Plan Implementation
EXPLANATION

a. Not Correct.
b. Not Correct.
c. Not Correct.
d. Correct. Notifications are required to be made 15 minutes after the declaration of an Emergency Classification.

*QNUM 076
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000015/017A211
QUESTION
The following Unit 1 plant conditions exist:

- A LOCA has occurred
- Command and Control has been transferred to the EOF
- The crew has transitioned to 1BFR-C.1, Response to Inadequate Core Cooling
- Containment pressure is 4 psig and stable
- CETC indicate 1250 degrees F and rising
- SG levels (%) are as follows:

1A	1B	1C	1D
0 NR	20 NR	0 NR	15 NR

- RCP #1 seal ΔPs are as follows:

	1A	1B	1C	1D
#1 seal ΔP (psid)	250	125	275	225

The crew is at step 17 in 1BFR-C.1 to check if RCPs should be started. The Unit RO recommends starting ONLY the 1D RCP to provide cooling to the core.

Which of the following is the correct response to the RO recommendation:

a. Direct the RO to start ONLY the 1D RCP.

b. Obtain authorization from the STA to start ONLY the 1D RCP.

c. Direct the RO to start the 1B and 1D RCP.

d. Obtain authorization from the EOF to start all RCPs.

ANSWER
c.
REFERENCE
1BFR-C.1, Response to Inadequate Core Cooling
FR-C.1 Background Information for WOG Emergency Response Guideline
BAP 1310-10, revision 10, HU-AA-104-101, Procedure Use and Adherence, Byron Addendum
EP-AA-112-100-F-01, Shift Emergency Director Checklist
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to determine and interpret the following as they apply to the Reactor Coolant Pump Malfunctions (Loss of RC Flow): When to jog RCPs during ICC
EXPLANATION
a. Is incorrect, because the procedure directs the starting of the 1B and 1D RCPs for the given conditions.

b. Is incorrect, because only the shift manager can allow deviations from directed procedure steps.

c. Is correct.

d. Is incorrect because ~~both 1B and 1D RCPs should be started.~~ because only the shift manager can allow deviations from directed procedure steps. Also, the EP procedures clarify that, if 10CFR50.54.x is invoked, decisions such as this must be made by an actively licensed SRO.

*QNUM 077
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000011 2.3.10
QUESTION

Unit Two experienced a Large Break LOCA and the operators are currently performing 2BEP ES-1.3, Transfer to Cold Leg Recirculation. Valve 2RH8716A, 2A RH Discharge Crosstie, will not close from the control room. An operator must be dispatched to the unit 2 penetration area to locally close the valve. An emergency dose of 10 Rem may be allowed if the individual volunteers and it is approved by the...

a. Station Emergency Director

b. Site Vice President

c. Health Physics Supervisor

d. Unit Supervisor

ANSWER
a.
REFERENCE
EP-AA-113, revision 8, Personnel Protective Actions

Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to perform Procedures to Reduce Excessive Levels of Radiation and Guard against Personnel Exposure.
EXPLANATION

a. Is correct. Only the Station Emergency Director may approve emergency dose above 5 Rem.

*QNUM 078
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000029A207
QUESTION
Unit 1 is operating at 100% power, all systems normally aligned and available. Reactor Trip Breaker (RTA/B) testing in progress with the following configuration:

- Reactor Trip Bypass Breaker A (BYA) and Reactor Trip Breaker B (RTB) are both RACKED IN and CLOSED
- Reactor Trip Bypass Breaker B (BYB) and Reactor Trip Breaker A (RTA) are both RACKED OUT and OPEN

The EO attempts to RACK IN and CLOSE Reactor Trip Bypass Breaker B (BYB) instead of Reactor Trip Breaker A (RTA). After manual closure, Reactor Trip Bypass Breaker B OPENs, and NO other breakers reposition.

What is the status of the reactor and what actions will the SRO direct?

a. tripped, direct the EO to locally open both the RTB and BYA breakers.

b. tripped, insert a manual reactor trip in accordance with 1BEP-0, Reactor Trip or Safety Injection Unit 1.

c. NOT tripped, IMMEDIATELY enter 1BFR-S.1, Response to Nuclear Power Generation / ATWS Unit 1.

d. NOT tripped, manually trip the reactor in accordance with 1BEP-0, Reactor Trip or Safety Injection Unit 1.

ANSWER
d.
REFERENCE
1BEP-0, Reactor Trip or Safety Injection Unit 1, Rev. 108; EF-2, ESF Setpoints, Rev. 0; BAR 1-10-B7, RX BYP BRKR 1A RACKED IN, Rev. 51; BAR 1-10-B8, RX BYP BRKR 1B RACKED IN, Rev. 51; TS 3.3.1, RTS Instrumentation, Condition N.
New
Higher
Ability to determine or interpret the RTB indicating lights as applied to an ATWS.
EXPLANATION

a. Reactor is NOT tripped.
b. Reactor is NOT tripped.
c. Reactor is tripped, but BFR-S.1 is entered after BEP-0
d. is correct.

*QNUM 079
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000038A208
QUESTION
Unit 2 has tripped due to a trip of the main generator and a subsequent Loss of Offsite Power. After the initial event the following adverse conditions occurred:

- The 2A Steam Generator has faulted and is at 200 psig.
- The transient caused by the faulted Steam Generator caused tube ruptures in ALL SGs (2A, 2B, 2C, and 2D SGs).

The Operations crew is at step 5 of 2BEP-3, SGTR, check ruptured SGs pressure.

Based upon the above conditions, which one of the following would be a CORRECT way to cool down the RCS?

a. Transition to 2BCA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery, and reduce temperature by releasing steam from the 2B, 2C, and 2D Steam Generators using the SG PORVs.

b. Transition to 2BCA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery, and reduce temperature by adjusting feed flow to the 2B, 2C, and 2D Steam Generators while maintaining SG PORVs closed.

c. Remain in 2BEP-3, Steam Generator Tube Rupture, and manually dump steam at the maximum rate from the 2B, 2C, and 2D Steam Generators using the SG PORVs.

d. Remain in 2BEP-3, Steam Generator Tube Rupture, and dump steam to the condenser from the 2B, 2C, and 2D Steam Generators.

ANSWER
a.
REFERENCE
2BCA-3.1, SGTR with Loss of Reactor Coolant – Subcooled Recovery Desired
2BEP-3, Steam Generator Tube Rupture
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to determine or interpret the following as they apply to a SGTR: Viable alternatives for placing plant in safe condition when condenser is not available.
EXPLANATION
a. Is correct.
b. Is incorrect because steaming the ruptured SGs is not an option in 2BEP-3.
c. Is incorrect because the transition to 2BCA-3.1 is required.
d. Is incorrect because adjusting feed flow to the Faulted SG is the other RCS cooldown method.

*QNUM 080
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000038 2.3.6
QUESTION
Given the following plant conditions on Unit 1:

- A Steam Generator Tube Rupture occurred.
- Secondary coolant specific activity is $1x10^{-5}$ μCi/gm dose equivalent I-131.
- The Fire and Oil Sump Discharge Rad Monitor, 0RE-PR005, was declared INOPERABLE.
- The Fire and Oil Sump Discharge Rad Monitor Normal/Bypass Switch is in the BYPASS Position.

Initially, liquid releases from the Fire and Oil Sump are allowed to continue if grab samples are analyzed for radioactivity every __(1)__ hours; if the Fire and Oil Sump Discharge Rad Monitor is NOT returned to service in __(2)__ days, releases must be terminated from this flowpath.

NOTE: TRM 3.11.a is attached.

	1	2
a.	12	14
b.	12	30
c.	24	14
d.	24	30

ANSWER
d.
REFERENCE
TRM 3.11.a, Radioactive Liquid Effluent Monitoring Instrumentation
BAR RM11-1-0PR05J
BRP 5820-13, Response to High Radiation Monitor Alarms
Bank
Higher
Proposed references to be provided to applicants during examination: TRM 3.11.aK/A
Knowledge of the Requirements for Reviewing and Approving Release Permits.
EXPLANATION
a. Not Correct.
b. Not Correct.
c. Not Correct
d. Correct. TRM 3.11.a, Radioactive Liquid Effluent Monitoring Instrumentation states the sample requirement is every 24 hours for effluents < 0.01 μCi/gm dose equivalent I-131 and it must be fixed within 30 days.

*QNUM 081
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000056 2.1.20
QUESTION
The following Unit 1 plant conditions exist:

- Unit 1 has experienced a Reactor Trip and Loss of Offsite Power.
- The crew is implementing 1BEP ES-0.1, Reactor Trip Response and 1BOA ELEC-4, Loss of Offsite Power.
- A safety injection then occurs.

The crew is expected to…

a. perform 1BOA ELEC-4 in parallel with 1BEP ES-0.1.

b. suspend performance of 1BOA ELEC-4 and implement 1BEP-0, Reactor Trip or Safety Injection.

c. suspend performance of 1BOA ELEC-4 and transition to 1BCA 0.0, Loss of All AC.

d. perform 1BOA ELEC-4 in parallel with 1BEP-0, Reactor Trip or Safety Injection.

ANSWER
b.
REFERENCE
1BOA ELEC-4, Loss of Offsite Power
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to execute procedure steps.
EXPLANATION
a. is incorrect, because 1BOA ELEC-4 should not be performed if an SI has occurred.
b. Is correct.
c. Is incorrect because there is no loss of all AC.
d. is incorrect, because 1BOA ELEC-4 should not be performed if an SI has occurred.

*QNUM 082
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 059 2.3.1
QUESTION
The amount of radionuclides stored in any ONSITE outdoor liquid storage tank that is NOT bermed is limited to that amount that…

a. would prevent exceeding regulatory limits to ground water if the tank's contents were uncontrollably released.

b. would NOT result in a radiation worker exceeding TEDE exposure if working 40 hours per week in close proximity to the tank.

c. can be directed through the plant radioactive waste system and released to the environment without exceeding National Pollution Discharge Elimination System (NPDES) limits.

d. will decay within 31 days to a regulatory limited level that can be safely released without further treatment.

ANSWER
a.
REFERENCE
T.S. 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program section C ensuring an uncontrolled release would result in concentrations below the limits of 10CFR20, appendix B, table 2, column 2.
New
Fundamental
K/A Knowledge of 10 CFR 20 and related facility radiation control requirements.
EXPLANATION
Distractor B based on limitations on the dose rate resulting from gaseous release beyond the site boundary stated in TS 5.5.4 section g.
Distractors D loosely based on TS 5.5.4, radioactive effluents controls program section f. that set limits on used of treatments systems to reduce releases when projected doses in 31 days would exceed 2% of the annual dose.
Distractor D references the NPDES permit which does not regulate radionuclides.

*QNUM 083
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000036A202
QUESTION
Unit 2 was in the process of moving fuel in the Reactor Building with all Refueling LCOs met when an earthquake occurred. The following conditions now exist:

- A fuel assembly was dropped in the refueling cavity.
- There has been a rapid drop in refueling cavity level.
- CNMT FH Incident Monitors 1RT-AR011 and 1RT-AR012 are reading greater than the high alarm.
- The CNMT Equipment Hatch is open.

Based upon the above, which one of the following conditions would result in exceeding offsite accident condition dose exposure limits?

a. If the refueling cavity water level drops to 24 feet above the reactor vessel flange.

b. The CNMT Equipment Hatch is NOT closed.

c. Both FHB Ventilation System trains do NOT operate.

d. The Containment Purge valves do NOT meet their 10 CFR 50, Appendix J, leakage criteria.

ANSWER
c.
REFERENCE
TS LCO 3.7.13, FHB Ventilation System
TS LCO 3.9.4, Containment Penetrations
TS LCO 3.9.7, Refueling Cavity Water Level
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to determine and interpret the following as they apply to the Fuel Handling Incidents: Occurrence of a fuel handling accident.
EXPLANATION
a. Is incorrect, because water level would need to fall below 23 feet.
b. Is incorrect, because even with the CNMT Equipment Hatch open, 10 CFR 50.67 limits would be met.
c. Is correct.
d. Is incorrect, because 10 CFR 50, Appendix J, leakage criteria are determined based upon a pressurized containment during a LOCA.

*QNUM 084
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 000069A201
QUESTION
Which of the following plant conditions would result in the Containment being INOPERABLE or NOT in the required status for the associated plant mode or plant configuration (assume that all other required systems/components for the associated plant mode are OPERABLE):

a. With Unit 1 in Mode 1, the interlock mechanism for the containment personnel air lock doors is found to be INOPERABLE.

b. With Unit 1 in Mode 1, the containment purge valve leakage rate is determined to be in excess of local leak rate requirements, but overall containment leak rate is within limits.

c. With Unit 1 in Mid-Loop operations, both doors of the containment personnel air lock are open.

d. With Unit 1 in Mode 6 and movement of recently irradiated fuel assemblies in progress, both doors of the containment emergency air lock are open.

ANSWER
d.
REFERENCE
TS LCO 3.6.1 and Bases, Containment
TS LCO 3.6.2 and Bases, Containment Airlocks
TS LCO 3.9.4 and Bases, Containment Penetrations
New
Fundamental
K/A Ability to determine and interpret the following as they apply to the Loss of Containment Integrity: Loss of Containment Integrity.
EXPLANATION
a. Is Incorrect, because the airlock interlock mechanism's operability is addressed separately under LCO 3.6.2.
b. Is incorrect, because containment isolation valves can exceed their limits without the containment being INOPERABLE. Overall containment leakage limits must be exceeded to make the containment INOPERABLE.
c. Is incorrect, because the Mode is inapplicable for LCO 3.6.1.
d. Is correct.

*QNUM 085
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A W/E13 2.1.32
QUESTION

Following a large break (LB) LOCA, a Red Path on the integrity CSF is received. The operators enter 1BFR-P.1, Response to Imminent Pressurized Thermal Shock Conditions, but transition back to 1BEP-1, Loss of Reactor or Secondary Coolant, after verifying RCS pressure and RHR flow.

Which of the following is the reason 1BFR-P.1 directs a “RETURN TO procedure and step in effect” during a LB LOCA?

a. The cooldown due to the LB LOCA will be of short duration. Once the vessel refill is complete, the downcomer will heatup relieving the Vessel Thermal stresses.

b. Due to backflow caused by RHR injection, the Cold Leg Temperatures are NOT a true indication of vessel cooldown rate and Pressurized Thermal Shock is NOT a concern.

c. The actions in 1BEP-1, Loss of Reactor or Secondary Coolant, will address mitigating the PTS concern.

d. Following a LB LOCA, the RCS CANNOT repressurize, therefore vessel integrity is NOT a concern.

ANSWER
d.
REFERENCE
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to explain and apply all system limits and precautions
PROVIDE EXPLANATION

*QNUM 086
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 006A210
QUESTION
With Unit 2 operating at 100% power, the NSO has provided the following information concerning the RWST to the Operations crew:

- RWST temperature is 38°F.
- RWST borated water level is 90 %.
- RWST boron concentration is 2260 ppm.

Based upon the above RWST parameters, what is the required action to take in accordance with LCO 3.5.4, Refueling Water Storage Tank, and, what is the bases for returning the parameter back to within its required Technical Specification values?

Restore the RWST to OPERABLE status due to…

a. Restore the RWST to OPERABLE status due to low water level to assure that following a LOCA the resulting sump pH will be maintained in an acceptable range.

b. Restore the RWST to OPERABLE status due to low RWST temperature. Returning the RWST temperature to within limits prevents ice blockage.

c. Restore the RWST to OPERABLE status due to low water level to ensure a sufficient supply of water is available post LOCA to support Containment Spray System pump operation.

d. Restore the RWST to OPERABLE status due to low boron concentration to assure that following a LOCA the resulting sump pH will be maintained in an acceptable range.

ANSWER
d.
REFERENCE
TS LCO and Bases for 3.5.4, RWST
New
Higher
Proposed references to be provided to applicants during examination: None
EXPLANATION
a. Is incorrect because RWST level is above 89% which is acceptable.
b. Is incorrect because RWST temperature is above 35 degrees.
c. Is incorrect because RWST level is above 89% which is acceptable.
d. Is correct. The reason for maintaining boron concentration for sump pH is dependent both upon RWST level and boron concentration. 2260 ppm is below the RWST limit, but it is above the SI accumulator limit.

*QNUM	087
Exam Date	2008/05/19
Reactor Type	PWR-WEC4
Exam Level	S
K/A	007 2.4.6

QUESTION

Unit 2 has experienced a total loss of Feedwater. The following conditions now exist:

- The Operations crew is performing 2BEP-0, Reactor Trip or Safety Injection.
- The Main Turbine did NOT trip automatically and had to be tripped locally.
- Containment Pressure is 0.5 psig.
- Both Unit 2 AF pumps have tripped and cannot be restored.
- SG wide range levels read as follows:

2A	2B	2C	2D
13%	10%	18%	12%

PZR pressure is 2250 psig.

Considering ONLY the above information, which ONE of the following would be a CORRECT action to respond to this event?

a. Transition to 2BFR-H.1, Response to Loss of Secondary Heat Sink, then establish RCS Bleed and Feed by actuating SI and bleeding through ONE PORVs to the PRT.

b. Transition to 2BFR-H.1, Response to Loss of Secondary Heat Sink, then establish RCS Bleed and Feed by actuating SI and bleeding through TWO PORVs to the PRT.

c. Transition to 2BEP-1, Loss of Reactor or Secondary Coolant, then transition to 2BEP-2, Faulted Steam Generator Isolation.

d. Transition to 2BEP-1, Loss of Reactor or Secondary Coolant, Check PZR PORVs and Isolation Valves to ensure at least ONE is operable/available.

ANSWER

b.

REFERENCE

2BEP-0, Reactor Trip or Safety Injection

2BEP-1, Loss of Reactor or Secondary Coolant

2BFR-H.1, Response to Loss of Secondary Heat Sink

New

Higher

Proposed references to be provided to applicants during examination: None

K/A Knowledge of symptom based EOP mitigation strategies.

EXPLANATION

a. Is incorrect, because 2BFR-H.1 requires to bleed thru 2 PORVs.
b. Is Correct.
c. Is incorrect, because the immediate transition is to 2BFR-H.1, Response to Loss of Secondary Heat Sink.
d. Is incorrect for the same reason as C.

*QNUM 088
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 010A201
QUESTION
In Modes 1, 2, and 3, LCO 3.4.9 requires that the pressurizer be OPERABLE.

For the Pressurizer to be OPERABLE, two groups of pressurizer heaters must be OPERABLE with the capacity of each group greater than 150 kW and capable of being powered from redundant Engineered Safety Features (ESF) power supplied buses.

Which one of the following events is the basis for this required power capability?

a. Steam Line Break.

b. Small Break LOCA.

c. Steam Generator Tube Rupture.

d. Loss of Offsite Power.

ANSWER
d.
REFERENCE
TS LCO and Bases 3.4.9, Pressurizer
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the PZR PCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Heater failures.
EXPLANATION
d. is the correct answer. As stated in the TS Bases, “Although the heaters are not specifically used in accident analysis, they provide the capability to maintain subcooling in the long term during loss of offsite power.”

*QNUM 089
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 026 2.2.22
QUESTION
Unit 2 is operating at 100% power. The 2A DG has been INOPERABLE for 24 hours due to planned maintenance. The Unit Supervisor has just declared the 2B containment spray pump as INOPERABLE due to a motor failure.

AT THIS TIME, and based upon the selections below, what is/are REQUIRED Technical Specification action(s) for this condition?

NOTE: TS LCOs 3.6.6 and 3.8.1 are attached.

1. Restore containment spray train B to OPERABLE status within 7 days.
2. Enter LCO 3.0.3 Immediately.
3. Be in MODE 3 within 6 hours.

a. 1 ONLY.

b. 1 AND 2 ONLY.

c. 2 ONLY.

d. 3 ONLY

ANSWER
a.
REFERENCE
TS LCO and Base 3.6.6, Containment Spray and Cooling System
TS LCO 3.8.1, AC Sources - Operating
New
Higher
Proposed references to be provided to applicants during examination:
TS LCOs 3.6.6 and 3.8.1
K/A Knowledge of limiting conditions for Operations and safety limits.
EXPLANATION

a. Is correct because the 2A containment spray pump does not have to be declared INOPERABLE for 4 hours under LCO 3.8.1, Action B.3.
b. Is incorrect because only the 2B containment spray pump is INOPERABLE at this time.
c. Is incorrect for the same reason as B.
d. Is incorrect because all AOTs are presently met.

*QNUM 090
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 103A203
QUESTION
Unit 1 is responding to a large-break LOCA. You have just transitioned to1BEP ES-1.3, Transfer to Cold Leg Recirculation, because of low water level in the RWST. The following occur in succession:

- SI reset is completed.
- The STA announces that a RED path exists due to containment pressure.
- You are unable to open the CC to RH heat exchanger valve 1CC9412A.
- Containment floor water level is 10 inches.
- Step 2 of 1BEP ES-1.3 checks CNMT floor water level – AT LEAST 8 INCHES (13 INCHES ADVERSE CNMT) with the RNO GO TO 1BCA-1.1, Loss of Emergency Coolant Recirculation

Based on the conditions listed above, you should...

a. Complete ALL steps in 1BEP ES-1.3 and then go to 1BFR-Z.1, Response to High Containment Pressure.

b. Align both RHR trains for recirculation and then go to 1BFR-Z.1, Response to High Containment Pressure,.

c. Immediately go to 1BFR-Z.1, Response to High Containment Pressure.

d. Go to 1BCA-1.1, Loss of Emergency Coolant Recirculation.

ANSWER
d.
REFERENCE
1BCA-1.1, Loss of Emergency Coolant Recirculation
1BFR-Z.1, Response to High Containment Pressure
Modified
Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to (a) predict the impacts of the following malfunctions or operations on the containment system and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Phase A and B isolation.
EXPLANATION
d. is correct, since Containment pressure is less than 13 inches for an adverse containment. Step 2 of 1BEP ES-1.3 transitions to 1BCA-1.1 based upon this condition.

*QNUM 091
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 001A209
QUESTION
The operators are performing BCA-0.1, Loss of All AC Recovery Without SI Required, step 3 (loading equipment on the ESF bus). The STA has just reported that CETCs are reading 720°F.

Which of the following is the correct response to the above conditions?

a. Immediately exit BCA-0.1 and proceed to BCA-0.2, Loss of All AC Recovery With SI Required.

b. Immediately exit BCA-0.1 and proceed to BFR-C.2, Response to Degraded Core Cooling.

c. Continue in BCA-0.1 until directed at step 5 to proceed to BCA-0.2, Loss of All AC Recovery With SI Required.

d. Continue in BCA-0.1 until directed to proceed to BFR-C.2, Response to Degraded Core Cooling.

ANSWER
a.
REFERENCE
BCA-0.1, Loss of All AC Power Recovery without SI Required
BCA-0.2, Loss of All AC Power Recovery with SI Required
BFR-C.2, Response to Degraded Core Cooling.

Higher
Proposed references to be provided to applicants during examination: None
K/A Ability to (a) predict the impacts of the following malfunction or operations on the CRDS – and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Station Blackout
EXPLANATION
a. Is correct
b. Is not correct because status trees are being monitored for information only.
c. Is not correct because the operator action summary pages directs a transition immediately when subcooling has been lost.
d. Is not correct because the operator action summary pages directs a transition immediately when subcooling has been lost.

*QNUM 092
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 011A204
QUESTION

Given the following conditions:

- The 1A CV pump is in operation
- The 1A CV pump is stronger than the 1B CV pump

What would be the consequences of swapping CV pumps and which Tech Spec(s) could be impacted? Assume 1CV121 remains in manual and total seal injection flow is 20 gpm following the swap.

a. Seal Injection flow would rise while charging flow would lower. Tech Specs 3.5.2, ECCS - Operating and 3.5.5, Seal Injection Flow could be impacted.

b. Pressurizer Level and Seal Injection Flow would both rise, and Tech Spec 3.5.2, ECCS – Operating could be impacted.

c. Letdown Temperature would rise and Tech Spec 3.5.5, Seal Injection Flow could be impacted.

d. Pressurizer Pressure would lower and no Tech Specs would be impacted.

ANSWER
c.
REFERENCE
Tech Spec. 3.5.2, ECCS – Operating
Tech Spec 3.5.5, Seal Injection Flow
1BOSR 5.5.1.1, RCS Seal Injection Flow Verification Monthly Surv.
New
Higher
Proposed references to be provided to applicants during examination: None

K/A Ability to (a) predict the impact of the following malfunctions or operations on the PZR LCS; and (b) based on those predictions, use procedures to correct, control, or mitigate the consequences of those malfunctions or operations: Loss of 1, or 2 charging pumps.

EXPLANATION

a. Not Correct. The seal injection flow rate being set with the stronger CV pump running would cause 1CV182 to be set further open than if the weak CV pump were used. Therefore, immediately following a CV pump swap seal injection flow would decrease and charging flow would decrease. Tech Spec 3.5.2 would not be impacted because both CV pumps would still be able to provide adequate ECCS injection flow.

b. Not Correct. Seal Injection Flow would lower.

c. Correct. Letdown temperature would rise as the result of lower charging flow through the regen hx and Seal Injection flow may not meet Tech Spec requirements.

d. Not Correct. Pressurizer Pressure would not be impacted by the decrease in charging flow.

*QNUM 093
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 035 2.4.21
QUESTION
The Heat Sink Critical Safety Function monitors ___(1)___ while it protects the radiological barriers of the ___(2)___.

	Monitors	Protects
a.	Wide range S/G Lvl, AF Flow, S/G Pressure	Fuel Clad & RCS Press boundary
b.	Narrow range S/G Lvl, AF Flow, S/G Pressure	Fuel Clad & Cnmt
c.	Wide range S/G Lvl, RCS Cold leg Temp, AF Flow	Cnmt & RCS Press boundary
d.	Narrow range S/G Lvl, AF Flow, S/G Pressure	Fuel Clad & RCS Press boundary

ANSWER
d.
REFERENCE
ST-1, Status Trees
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the parameters and logic used to assess the status of Safety Functions.
EXPLANATION
a. Wide Range S/G Level is incorrect
b. Cnmt is incorrect
c. Wide Range S/G Level & RCS Cold Leg Temp are incorrect
d. Correct, The Heat Sink Critical Safety Function uses Narrow Range S/G Level, AF Flow, & S/G Pressure to determine if the Fuel Matrix and RCS Pressure Boundary are protected.

*QNUM 094
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.1.4
QUESTION
Which one of the following is the MINIMUM requirement for the total number of on-shift Fire Brigade members?

a. Five (5), including the Field Supervisor as Fire Chief.

b. Six (6), including the Field Supervisor as Fire Chief.

c. Five (5), NOT including the Field Supervisor as Fire Chief.

d. Four (4), NOT including the Shift Technical Advisor as Fire Chief.

ANSWER
a.
REFERENCE
BAP 1210-

Fundamental
Explanation
a. Is correct as stated in the procedure.

*QNUM 095
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.1.20
QUESTION
The Station has experienced a large break Loss of Coolant Accident on Unit 2. The Shift Manager has assumed the duties of the Shift Emergency Director and is in Command and Control.

Which of the following is a list of the Shift Emergency Director's Non-delegable responsibilities?

a. Classification of the Emergency
Notification of the Site Vice President
Notification of the State and Federal Agencies
Site Assembly / Accountability

b. Classification of the Emergency
Authorization for Emergency Dose Exposure
Notification of the State and Federal Agencies
Determination of Protective Action Recommendations to the State

c. Classification of the Emergency
Authorization for Emergency Dose Exposure
Site Assembly / Accountability
Determination of Protective Action Recommendations to the State

d. Classification of the Emergency
Notification of the Site Vice President
Notification of State and Federal Agencies
Determination of Protective Actions for Plant Personnel

ANSWER
b.
REFERENCE
EP-AA-11,
EP-AA113-F-02, Authorization for Emergency Exposure
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of General Operating Crew Responsibilities during Emergency Operations.
EXPLANATION
a. Not Correct. Shift Emergency Director may delegate notification of Site VP & Site Assembly.
b. Correct. These are the four non-delegable duties of the Emergency Director.
c. Not Correct. Shift Emergency Director may delegate Site Assembly.
d. Not Correct. Shift Emergency Director may delegate notification of Site VP & doesn't make Protective Action Recommendations to plant personnel.

*QNUM 096
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.2.5
QUESTION
Which of the following conditions would require a 50.59 Evaluation to be performed?

a. The Emergency Planning (EP) Coordinator is changing the Protective Action Recommendations (PARS) in the Emergency Plan.

b. Mechanical Maintenance will be performing a like-for-like replacement of the 1B SI pump rotating element.

c. System Engineering will be adversely changing the overall isolation time response of several containment isolation valves in the Containment Chiller Water System.

d. The Station Fire Chief is making changes to the Station Fire Plan that is going to change the temperature at which sprinkler heads actuate in the Turbine Lube Oil Area.

ANSWER
c.
REFERENCE
LS-AA-104-1000, 50.59 Resource Manual
LS-AA-128, RegulatoryReview of Proposed Changes to the Approved Fire Protection Program
New
Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the process for making changes in the Facility as Described in the Safety Analysis Report.
EXPLANATION
a. Not Correct. Emergency Plan changes are evaluated under 10 CFR 50.54(q)
b. Not Correct. Maintenance Activities are covered by 10 CFR 50.65
c. Correct. A change to the containment isolation function is evaluated in the UFSAR and thus this change needs to be evaluated using the 10 CFR 50.59 process.
d. Not Correct. Fire Plan changes that due not alter safe shutdown analysis are evaluated using LS-AA-128.

*QNUM 097
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.2.18
QUESTION

A MMD FLS and Plant Engineer are reviewing a troubleshooting plan with the U-1 US. Per MA-AA-716-004, Conduct of Troubleshooting, Operations Department will review Attachment 1, Troubleshooting Log, to ensure…

a. Engineering Department has performed a Risk Assessment of the troubleshooting activities.
b. equipment calibrations are within current calibration frequency
c. Plant Engineering review of troubleshooting results is assigned to the proper system engineer.
b. adequate bounds on the troubleshooting activities have been established to limit plant impact.

ANSWER
d.
REFERENCE
MA-AA-716-004, Conduct of Troubleshooting sections 3.3.4
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the Process for Managing Troubleshooting Activities.
EXPLANATION
a. Need to provide explanations
COMMENTS

*QNUM 098
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.3.5
QUESTION
Given the following plant conditions:
U-1 was at 100% power
The main turbine tripped due to an EHC leak
NEITHER Reactor Trip Breaker opened automatically or manually from the Main Control Room.

The SRO directs the use of 1BFR S.1, Response to Nuclear Power Generation/ATWS, in order to …

a. mitigate the severe RCS pressure transient.

b. mitigate the severe RCS temperature rise.

c. adequate borating to shutdown the reactor within 10 minutes.

d. mitigate a possible loss of feedwater after Safety Injection is actuated to trip the reactor.

ANSWER
a.
REFERENCE

Higher
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the Use and Function of Personnel Monitoring Equipment
PROVIDE EXPLANATION
a. Need to provide explanations

*QNUM 099
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.3.7
QUESTION
The following plant conditions exist:

- Workers are making repairs to the 1B Centrifugal Charging Pump.
- A Job Specific RWP was issued at the start of the job.

Which one of the following DOES NOT require the Job Specific RWP to be placed into a "HOLD" status?

a. Maintenance personnel notice a significant increase in radiological conditions.

b. Maintenance First Line Supervisor wants to use different tools than those already at the pump to disassemble the pumps.

c. Radiation Protection personnel observe Maintenance personnel not complying with RWP requirements.

d. Operations will be conducting an LLRT in the area which could affect radiological conditions in the 1B CV Pump Room.

ANSWER
b.
REFERENCE
RP-MW-403-1001, Radiation Work Permit Processing
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of the Process for Preparing a Radiation Work Permit
EXPLANATION
a. Not Correct. This is a condition that requires the RWP to be placed on HOLD.
b. Correct. The Maint. FLS can not place the RWP on hold.
c. Not Correct. This is a condition that requires the RWP to be placed on HOLD.
d. Not Correct. This is a condition that requires the RWP to be placed on HOLD.
COMMENTS

*QNUM 100
Exam Date 2008/05/19
Reactor Type PWR-WEC4
Exam Level S
K/A 2.4.7
QUESTION
The following plant conditions exist:

- Loss of All AC Power occurred 60 minutes ago

While implementing 1BCA 0.0, Loss of All AC Power, the operators are performing Step 55, SELECT PROPER RECOVERY PROCEDURE. The following conditions exist:

- RCS subcooling is ACCEPTABLE.
- PZR level is at 22% with containment pressure at 8 psig.
- No SI pumps are running.

The recovery procedure to be entered, and the reason for selection, is…

a. 1BCA 0.2, Loss of All AC Power Recovery With SI Required, because PZR level is greater than 20% with an ADVERSE containment.

b. 1BCA 0.1, Loss of All AC Power Recovery Without SI Required, because SI equipment has not actuated.

c. 1BCA 0.1, Loss of All AC Power Recovery Without SI Required, because PZR level is greater than 12% with an ADVERSE containment.

d. 1BCA-0.2, Loss of All AC Power Recovery With SI Required, because PZR level is less than or equal to 28% with an ADVERSE containment.

ANSWER
d.
REFERENCE
1BCA-0.0, Loss of all AC Power
New
Fundamental
Proposed references to be provided to applicants during examination: None
K/A Knowledge of Event Based EOP Mitigation Strategies
EXPLANATION

a. Is incorrect because the criteria for transition to BCA 0.2 is that PZR level is less than or equal to 28% with an adverse containment. Low PZR level is the concern not high PZR level.
b. Is incorrect. SI pumps not running would transition to BCA-0.1.
c. Is incorrect for the same reason as A.
d. Is correct.